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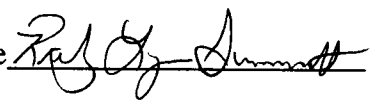
Signature 

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ASSESSMENT OF THE CONTAINMENT RESPONSE OF AN ADVANCED PWR TO SELECTED SEVERE ACCIDENT SEQUENCES

A Thesis Presented

for the Master of Science Degree

The University of Tennessee, Knoxville

Ricky Lynn Summitt

May 1994

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DEDICATION

I wish to dedicate this thesis to memory of

Dr. Petro F. Pasqua

for having given me the privilege of knowing him
during his life and to my former manager

Dr. Mario H. Fontana

for all that he taught me about being an engineer.

ACKNOWLEDGMENTS

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ABSTRACT

A limited severe accident assessment of a postulated advanced pressurized water reactor (APWR) design is performed and is based, in part, on requirements provided by the Electric Power Research Institute Advanced Light Water Reactor Utility Requirements Document. The work includes the installation of the Modular Accident Analysis Program (MAAP) on a personal computer, development of a summary screen, benchmarking the MAAP code to baseline assessments, the development of a pseudo-APWR model, and an assessment of the containment response of the APWR to selected severe accident sequences. The results of this assessment are then compared to those provided for similar current generation plants.

The benchmark exercise was performed successfully and no significant variance in the major parameters was identified. Based on the severe accident assessment, the overall conclusion drawn from the assessment is that the APWR design offers additional safety margin for mitigating severe accidents. The main improvement is the plant response to high pressure melt ejection (HPME). Improvements are also noted in precluding basemat failure and late containment overpressurization.

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Abbreviations and Acroynms

ALWR - Advanced Light Water Reactor

APWR - Advanced Pressurized Water Reactor

ARSAP - Advanced Reactor Severe Accident Program

BWR - Boiling Water Reactor

CDC - Control Data Corporation

DCH - Direct Containment Heating

EPRI - Electric Power Research Institute

HPME - High Pressure Melt Ejection

IPE - Individual Plant Examination

IRWST - In-Containment Refueling Water Storage Tank

LOCA - Loss of Coolant Accident

MAAP - Modular Accident Analysis Program

NRC - Nuclear Regulatory Commission

PC - Personal Computer

PORV - Power Operated Relief Valve

PRA - Probabilistic Risk Assessment

PWR - Pressurized Water Reactor

RCP - Reactor Coolant Pump

RCS - Reactor Coolant System

RWST - Refueling Water Storage Tank

SDV - Safety Depressurization Valve

SRV - Safety Relief Valve

TMI - Three Mile Island

USDOE - U.S. Department of Energy

CHAPTER 1

INTRODUCTION

This document is divided into five sections. Section 1 provides a summary introduction to the ALWR program and the work documented by this thesis. Section 2 addresses the modification and installation of the MAAP code on the PC and the benchmark exercise. Section 3 examines the ALWR design and the severe accident progression analysis. The results of the analysis are described in Section 4. Conclusions and recommended follow-on activities are discussed in Section 5.

1.1 Advanced Light Water Reactor Program

The United States nuclear utilities and vendors, with assistance from the United States Department of Energy (USDOE), are currently developing the designs for the next generation of nuclear power plants. As a part of this overall effort, the Electric Power Research Institute (EPRI) is developing the Utility ALWR Requirements Document.[1] The Requirements Document effort centers around the development of a set of technical design requirements for these plants. These requirements address all aspects of plant design and reflect the vast experience gained from the operation of present generation reactors.

The Requirements Document is comprised of three volumes. The first volume sets forth policy statements which address issues such as simplification, safety, and availability. This volume sets overall top-tier requirements which serve as the basis for the technical requirements which follow. Volume two addresses both top-tier and

technical requirements for the evolutionary plant designs. Evolutionary designs are defined as those which build on present designs and represent an evolution in design which accounts for lessons learned. Volume three provides the top-tier and technical requirements for the passive plant designs. The passive plant designs incorporate many passive safety features not present in existing plants and as such represent a revolutionary concept. Volumes two and three are further divided into thirteen chapters which address certain aspects of the design, i.e., Engineered Safety Features, Plant Cooling Water Systems, etc.,.

It is these detailed requirements which provide guidance for the design of the next generation reactors. The guidance provided may require that a specific design aspect be addressed in a particular fashion (e.g., how many independent trains of safety injection) or provide an overall objective (e.g., provide a reliability assurance program) which the designer must accomplish using a method of the designer's choice. Using these requirements, information provided from the reactor vendors, and standard design practices, the major design characteristics of an ALWR design which fulfills the intent of the Requirements Document can be determined. It is this process which allows evaluation of the design requirements to ensure that the top-tier design objectives can be accomplished.

One of the top-tier requirements for the evolutionary plant design deals with the ability of the ALWR to mitigate severe accidents and minimize public radiation exposure. This requirement specifies that:[1]

> *"Severe accident risk shall be evaluated using the PRA, and it shall be demonstrated by the Plant Designer that the cumulative frequency is less that 1x10-6 per reactor year for sequences resulting in greater than 25 rem whole body dose over 24 hours at .5-mile from any individual reactor."*

The purpose of this requirement is to assure that the ALWR design provides a significant margin beyond current United States Nuclear Regulatory Commission (NRC) safety goals[2] and to support the increased public acceptance of nuclear power. Additionally, the NRC severe accident policy statement states that "... new standard plant will achieve a higher standard of severe accident performance than...prior designs". For these reasons, a specific top-tier requirement is provided which outlines the desired level of safety.

The requirement to minimize public radiation exposure does not assure that this level of protection will be provided by the design requirements. To provide a level of confidence that the level of safety required by the Requirements Document can be achieved, evaluation of a plausible plant design based on the requirements is needed. This assessment can be performed using PRA techniques combined with deterministic accident progression models. The work performed in support of this thesis provides a limited deterministic assessment of the responses of a plausible ALWR design and of a current generation design to selected severe accident sequences. A comparison of the results for the two designs can be used as a qualitative measure of the effectiveness of the technical requirements in the Requirements Document in arriving at a design which can achieve this important top-tier safety requirement.

1.2 Survey of Previous Work

The use of PRA and deterministic analyses to determine the ability of present designs to successfully respond to severe accident sequences is an ongoing process which may be best illustrated by the Industry Degraded Core Rulemaking (IDCOR) Program[3,4] and the Individual Plant Evaluation[5] (IPE) . These efforts are the result of industry and NRC efforts to perform a detailed evaluation of plant response to severe accidents.

The industry program, IDCOR, involved all major reactor vendors, nuclear utilities, and technical consultants and resulted in an understanding of severe accidents and the performance of safety systems and the containment during beyond design basis conditions. As a result of this work, most of the literally hundreds of severe accident issues identified during interactions with the NRC were resolved and the NRC concluded that present reactors are safe. An important product from the IDCOR program is the Modular Accident Analysis Program[6] (MAAP). This computer model uses the results of experiments and first-order models for physical processes to predict plant characteristics during a severe accident. A major benefit of the MAAP code is the ability to model the plant response in an integrated fashion. Until the development of the MAAP code, most accident analyses were performed using several separate computer programs which did not have the ability to interact and take into account cause-and-effect relationships. The lack of an integrated code required that the analyst guess at the effect of the various relationships. The development of the MAAP code resulted in the ability to investigate severe accident phenomena and containment response in an integrated fashion.

Although the IDCOR program was successful in assessing different vendor designs, the analysis highlighted the variety of plant design configurations present in the United States. Thus, although the general conclusion could be reached that the present plants are safe, a review of each plant design was considered important to confirm this general conclusion. The IPE process served as the means by which each plant was evaluated. The IPE evaluated the plant response to severe accidents for the purpose of identifying any plants which, due to plant-specific reasons, were outside the range of acceptable severe accident mitigation performance. Probabilistic modeling techniques are used to identify the most likely accident sequences and, for the vast majority of IPEs, the MAAP code is used to evaluate containment response.

Similar analyses have been conducted on the ALWR designs. The measure most commonly used to assess the ability of a plant which is designed in accordance with the Requirements Document to meet the risk requirement is the frequency of a release from an impaired containment. If the frequency from all contributing sequences is less than the requirement frequency, then the plant design would meet the requirement.

The U.S. Department of Energy sponsored the Advanced Reactor Severe Accident Program[7] (ARSAP) to assist EPRI in the development of the Requirements Document. As a part of this effort, ARSAP developed a PRA for a plant design which meets the Requirements Document for an advanced PWR design.[8,9] The PRA calculated the expected frequency of release for an evolutionary PWR design and found that the frequency of release was less than the target value. ABB/CE has developed a level 2 PRA for its evolutionary plant design, the System 80+.[10] The results of this PRA are still being refined. The initial results, however, concurred with the results of the referenced ARSAP report.

Part of the goal of the Requirements Document top-tier requirement is to highlight the improved safety of the evolutionary plant design over present plant designs. ARSAP developed several topical papers[11] and reports[12,13] which dealt with the important severe accident issues and identified how the ALWR designs represented an improvement in severe accident mitigation. All of these efforts provided support for and assurance that the requirements contained in the Requirements Document would result in a design which will meet the safety objective described above.

1.3 Overview of the Models in MAAP

The MAAP code is written in a modular format which allows specific models to be examined and updated without impacting others. Data is passed between the specific models by the use of large common blocks which contain the bulk of the data required.

The module hierarchy has four different levels. At the highest level are the executive subroutines. These routines pass variables to the other subroutines, control input and output, and control integration.

At the next level are the system and region subroutines. These subroutines are used to simulate plant operation. Operation of the safety injection system is an example of the functions performed by this type of subroutine. The center of the MAAP code is the phenomenological subroutines. These subroutines track the phenomena as they occur within the many regions of the RCS and the containment. Hydrogen burning and radionuclide transport are two examples. The last group contains the important property library subroutines. These routines provide necessary physical information to the other subroutines.

The MAAP code contains coupled first-order ordinary differential equations with time being the independent variable. MAAP contains on the order of 3000 separate dependent variables. The majority of equations address the conservation of mass or energy within a defined volume and assume that no rapid changes occur in any parameter during a time step in order to reduce the mathematical complexity. As examples, the mass and energy equations are similar to those shown below:

$$\dot{m} = W_{in} - W_{out} + Sources - Sinks$$

$$\dot{U} = W_{in}h_{in} - W_{out}h_{out} + Q_{gain} - Q_{loss}$$

$$\dot{T} = [HA(\Delta T) + Q] / mc_p$$

where:
$\dot{m}$ = rate of change of mass in the volume (kg/s)
W_{in}, W_{out} = mass flow rate in and out of volume (kg/s)
Sources, Sinks = mass flow from additional sources or losses of mass within the volume (kg/s)
$\dot{U}$ = rate of change of internal energy in the volume (J/sec)
h_{in}, h_{out} = specific enthalpy of masses entering and exiting the volume (J/kg)
Q, Q_{gain}, Q_{loss} = heat, heat gains or losses due to sources and sinks in the volume (J/sec)
$\dot{T}$ = rate of change of temperature in the volume (K/sec)
H = heat transfer coefficient (J/s-m^2-K)
A = heat transfer area (m^2)
ΔT = Temperature difference between the volume and it surroundings (K)
m = mass (kg)
c_p = specific heat with constant pressure (J/kg-K)

The first example equation addresses the conservation of mass within a volume. The first two terms address mass changes as a result of interactions between different volumes. The "source" and "sink" terms represent potential sources and sinks found in the volume. Condensation and evaporation within the volume are examples of a source and a sink considered when addressing the mass of water present in the volume.

The equation for internal energy provides an insight into how the MAAP equations are simplified. The equation accounts for changes in enthalpy and other heat sources but does not account for any *pv* terms in the determination of the change in internal energy. The basis for this exclusion is that the change in energy for a volume is dominated by other sources (hydrogen burning, decay heat, etc.) when considering severe accidents and this term has very little impact on the result.

The MAAP code integration process is illustrated in Figure 1. The physical models are contained in the phenomenological and thermodynamic routines. The integration routine uses the rate of change information to update the variables between time steps.

The models in MAAP can be considered to be a member of one of four main groupings. These are the containment, primary system, safety systems, and phenomenology. A brief description of each as it applies to a PWR is provided below.

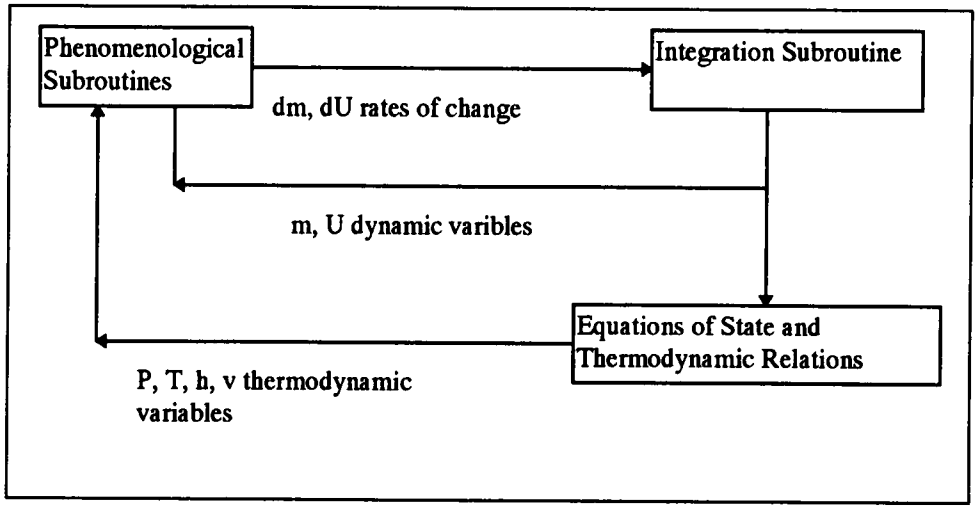

Figure 1. MAAP Integration Scheme (Reproduced from Reference 6)

The containment model is based on a nodal modeling approach which maintains a balance between the individual volumes. Important parameters (e.g., steam, water, hydrogen, other gases) are transported between the different volumes based on physical laws and conditions within the different volumes. The four major containment volumes are the upper compartment, lower compartment, annulus, and cavity. Other volumes are the RCS, pressurizer, quench tank, and the annular compartment. The MAAP containment model also includes provisions for the containment fan coolers.

The safety system models address important systems which influence the progression of the accident sequence. Safety systems (e.g., safety injection, containment spray, accumulators) and some non-safety systems (e.g., normal RCS makeup) are addressed.

The event flags provided in the MAAP sequence file control how the systems will be applied for a specific sequence. By setting the different event flags a modeled system can be set for automatic operation, manual actuation, or the system can be failed.

The primary system model addresses the RCS by subdividing the RCS into subregions. Flow equations are provided for all subregions and for the external sources (e.g., safety injection) and sinks (e.g., RCS pipe break). The flow equations track water, gases, and radionuclides. The model also provides the interface between the containment and the RCS. Examples of the phenomenology addressed in the MAAP models are provided in Table 1.

Table 1
Examples of Phenomena Addressed in MAAP

RCS heat transport
RCS loss of water inventory and accumulation in containment
Emergency core cooling sources
Core uncovery, heatup, hydrogen, and fission product release in the RCS
Hydrogen migration into containment
Hydrogen and carbon monoxide combustion in the containment

Over 300 subroutines are provided in MAAP to address required modeling. MAAP has been compared to separate-effects tests, integral experiments, and actual plant experience. Correlation of the MAAP predicted results to these sources provides confidence in the individual models present in MAAP.[6]

In order to gain insights into the underlying model development, the development of a section of one module is provided here. The selected module addresses interactions within the core of uncovered fuel nodes. Four calculations are performed within this module.

- Steam generation within the RCS due to quenching
- Fuel pin geometry for each node
- Distribution of steam and hydrogen flow rates
- Cladding oxidation and heat transfer rate

The development of the cladding oxidation model which is used to predict hydrogen generation from in-vessel sources is selected for further discussion. The oxidation reaction is shown below.

$$Zr + 2H_2O \Rightarrow ZrO_2 + 2H_2 + heat$$

The rate of hydrogen generation is directly related to the oxidation of zirconium by this reaction. The oxidation rate is divided into two regimes that are based on temperature. Prior analysis has found that the rate of oxidation increases after the zirconium oxidation changes from endothermic to exothermic. This occurs when the zirconium temperature reaches 1850K. Two empirical correlations[14,15] that are based on experimental data are used to determine the rate of change in fuel pin thickness.

$$\dot{x} = \frac{C_1}{2\rho_{Zr}^2 x} e^{\frac{-k1}{RT_{pin}}} \qquad T_{pin} < 1850\ K$$

$$\dot{x} = \frac{C_2}{2\rho_{Zr}^2 x} e^{\frac{-k2}{RT_{pin}}} \qquad T_{pin} > 1850\ K$$

where:
$\dot{x}$ = rate of change in fuel pin thickness due to zirconium oxidation (cm/s)
C_1 = 294 (kg^2/cm^5)
C_2 = 3330 (kg^2/cm^5)
k1 = 1.654×10^8 (J/kg)
k2 = 1.884×10^8 (J/kg)
x = fuel pin thickness (cm)
R = ideal gas constant (J/kg-K)
ρ_{Zr} = density of zirconium (kg/cm^3)
T_{pin} = temperature of fuel pin (K)

The predicted rate of change is integrated over the time step to obtain an average rate of change. The oxidation rate (in terms of thickness oxidized during the time step) can then be used to calculate the molar oxidation rate using the following equation:

$$\dot{n}_{Zr} = \frac{\dot{x}\, A\, \rho_{Zr}}{M_{Zr}}$$

where:
$\dot{n}_{Zr}$ = molar oxidation rate (mole/s)
$\dot{x}$ = rate of change in fuel pin thickness due to zirconium oxidation (cm/s)
A = area of fuel pin (cm^2)
ρ_{Zr} = density of zirconium (kg/cm^3)
M_{Zr} = molar weight of zirconium (kg/mole)

One additional change is made to account for physical limits present within an actual core. In order for oxidation to occur, both steam and unoxidized zirconium must be present. Thus, the amount of zirconium oxidized is limited by the smaller quantity of steam and zirconium. If sufficient steam and zirconium are present the oxidation reaction is calculated as indicated above. If insufficient steam or zirconium is available to complete the oxidation, the model limits the oxidation to reflect a deficiency in the quantity of material available. In other words, the model terminates the oxidation reaction if either the available steam or zirconium is exhausted. The molar rate of hydrogen production is found by doubling the molar rate of zirconium oxidation.

It is important to recognize that the state of knowledge of severe reactor accidents is constantly changing and that methods used in the modeling severe accident phenomena are subject to change. The modular format of MAAP allows for the physical models to be easily adapted for different hypotheses and plant configurations.

1.4 Description and Scope of Thesis Project

As stated in the prior section, one goal of the EPRI safety requirement is to show how the evolutionary plant designs are more robust with regard to severe accidents than present plants. To support this assertion, the project documented by this thesis is a comparison of containment responses to severe accidents for a current generation pressurized water reactor (PWR) plant design and a postulated advanced PWR design. The analysis examines the plant response for three of the key issues related to severe accidents and is performed using the more likely accident sequences identified for the advanced PWR.

In order to perform the comparison, changes to the MAAP code were required to allow it to be executed on an MS-DOS compatible personal computer (PC). The conversion of MAAP to the PC is one of three distinct phases of this thesis project. The first phase involves the installation of Revision 11 of the Modular Accident Analysis Program (MAAP) onto an MS-DOS compatible personal computer (PC). The MAAP code is written in FORTRAN for execution on a Digital Equipment Corporation VAX platform. The MAAP code was adjusted for the PC, loaded onto a PC, and compiled. A benchmark exercise was performed using the sample problem provided with the MAAP code. Selected results from the PC-compiled program were compared to the VAX-generated results provided in the user's manual. Correlation of these results served as a benchmark of the PC version of MAAP.

The second phase of the project involves developing a PC screen interface for the MAAP code. The screen interface provides the analyst with real-time updates of important parameters. These parameters are displayed on the PC screen in a structured format which allows on-line monitoring of analyses to gain insights into severe accident phenomena.

Finally, the third and major phase of the project involves a limited comparison of severe accident responses between a present PWR design, which is represented by a Zion-like plant, and an advanced PWR design as defined by the Utility ALWR Requirements Document. In this assessment specific attention is given to the response of the containment to ex-vessel debris cooling, high pressure melt ejection, and hydrogen concentration. The parameters in these three phenomena represent the most uncertain parameters associated with severe accident progression and can alter overall conclusions significantly.[16]

As a final note, it is important to remember that the analysis is based on the information contained in the EPRI Requirements Document and as such does not represent any particular advanced PWR design. Due to the lack of design detail, a number of assumptions were made which could impact the results of the analysis. In many cases, plant parameters used in this analysis represent conservative estimates which could bias a particular set of parameters. The overall results of the analysis, however, should not be affected.

CHAPTER 2

CONVERSION OF MAAP TO A PERSONAL COMPUTER

This section describes the modification and installation of the MAAP code on a PC. Modifications to the source code, the development of the screen display, and the bench marking exercise are discussed.

2.1 Development of MAAP Display Screen

The MAAP code was developed for batch mode execution on a central computer which allows the user to perform other tasks during execution. As such, the analyst does not receive much information on the status of a particular assessment until the run is completed. The use of a desktop as a dedicated machine provides an opportunity to improve the user interface by providing a graphical output which provides the user with real-time information about plant status as being predicted by the MAAP code.

A personal computer screen interface for MAAP is considered an important added feature to support the work performed for this thesis. A FORTRAN subroutine was developed to allow MAAP users the ability to observe the status of selected MAAP values during the execution of the MAAP code on an IBM-compatible personal computer. The interface provides a real-time presentation of selected MAAP variables which are considered of interest. These variables are graphically grouped into larger categories to create a better user interface and to allow the user to concentrate on specific aspects of the ongoing analysis. The variables were selected based on previous studies and other documentation and reflect the majority of important severe accident parameters which are used to assess containment performance.

The graphical screen routine is written in standard F77 commands and is presented in Appendix A. The routine is integrated into the MAAP program. The screen routine, however, is self-contained and is independent of other MAAP subroutines. Independence, however, does not preclude the screen program from possibly affecting the operation of the MAAP code if it is installed improperly. The following text outlines the interface development.

The first step in defining the graphical subroutine is to develop an easy interface between the subroutine and the main program that allows variables to be shared. The MAAP code, as do many FORTRAN codes which manipulate large amounts of data, utilizes common block statements which can be used to group data. It also allows for data to be transferred in bulk quantities between subroutines. The graphical subroutine utilizes this information in order to obtain the values of the selected parameters.

Care was taken in the choice of a common block format. If the variables in the subroutine common block file do not form a one-to-one match with the equivalent variables found in the other MAAP block files, the potential exists for variable values to be altered.

To preclude this problem a simple solution was employed. MAAP common block statements from the MAAP main subroutine were examined and a particular common block identified which contained the variable of interest.

Once the common block was added to the graphical subroutine, the next step was to provide a means in the MAAP code to call the screen program. This was accomplished by adding a few steps to the MAAP program within the MAIN subroutine. The MAIN subroutine of the MAAP code is the controller which directs the high-level execution of the MAAP code. It provides calls to the other routines, such as data output.

Before the screen routine was developed the most important severe accident parameters displayed by the MAAP code were identified. Since there is limited space for displaying information only the most important characteristics were chosen for display. The PWR version of the MAAP code contains thousands of variables and over 200 user and computer controlled flags. The flags are set to identify the status of equipment, change physical models, and to highlight important events in the accident progression.

Using the insights gained from examing prior studies, 22 different variables and 17 flags events were chosen to be displayed by the screen subroutine. The variables are provided in eight logical groups which center around plant system and physical boundaries and are presented in British units. The parameters allocated to each group are described in Table 2. The variables are representative of the kinds of information most useful in determining success criteria and containment performance without performing a detailed assessment of the results.

In addition to the listing of specific variables the subroutine provides a visual status of several flags contained in the MAAP code. Event code flags are used to identify the occurrence of important events in the accident progression and to determine the status of important safety systems. By monitoring the status of these flags, the status of systems and the existence of certain phenomena are determined. The status of the event flag is presented by either a normal background or a highlighted (orange) background. The highlighted background indicates that an event has occurred, for example, a PORV opening. A lack of highlight means that the event is not currently in effect.

Table 2
Screen Variable Groupings

Group Title	Variables and Descriptions
Core	Hottest core node: (TCRHOT) - This variable represents the temperature of the hottest core node.
Reactor Vessel (RV)	RV water temperature (TWCR) - represents the bulk water temperature in the core region.
	RV gas temperature (TGCR) - represents the bulk gas temperature in the core region.
	RV total water mass (MWCR+MWBH+MWUH) - the three values provide an estimate of the total amount of water in the RV.
	RV water level (ZWV) - this represents the collapsed water level in the RV.
Primary System (PS)	PS pressure (PPS) - provides an estimate of primary system pressure.
	PS temperature (TWPS) - estimate of the bulk primary system water temperature.
	PS hydrogen mass (MH2CRT) - estimate of the amount of hydrogen present in the core region of the primary system.
Pressurizer (PZ)	PZ temperature (TWPZ) - provides the pressurizer water temperature.
	PZ pressure (PPZ) - the pressurizer pressure.
	PZ water level (ZWPZ) - the height of the water in the pressurizer.
	PZ hydrogen (MH2PZ) - the quantity of hydrogen in the pressurizer.
Safety Injection (SI)	SI flow (WESFDC+WESFCL+WESFHL) - the total SI flow (including accumulators) into the primary system.
	RWST level (XCRLS) - the percent level of the RWST based on MAAP parameters ZWRWST, MRWST0, and ARWST. This value is calculated by PCSI.
Steam Generator (SG)	Unfaulted SG water mass (MWUS) - the mass of water present in the unfaulted SG.
	Faulted SG water mass (MWBS) - the mass of water present in the faulted SG.
Feedwater	Feedwater rate (WWFWUS) - the feedwater addition rate to the unfaulted SG.
	Feedwater rate (WWFWBS) - the feedwater addition rate to the faulted SG.
Containment	UC pressure (PA) - represents the pressure in the upper compartment (compartment A).
	Noble gases in the containment (QGFPA through QGFPU) - the sum of the noble gases estimated for the different containment compartments.
	UC hydrogen (MH2A) - the mass of hydrogen present in the upper compartment.
	Cavity hydrogen (MH2C) - the mass of hydrogen estimated to be present in the cavity.

In addition to highlighting, certain phenomena are also reflected by a change in the indicator text. For example, when the safety injection system switches from the RWST source to the containment sump, the status indicator highlights the indicator and changes it from RWST to RECIRC in order to indicate recirculation is underway.

Similar changes are found for the condensate storage tank indicator (CST AVAIL, CST EMPTY) and the status of containment integrity (UC ISOLT, UC FAILED; LC ISOLT, LC FAILED). A complete listing of the event flags monitored by the PCSI routine is presented in Table 3. A sample screen is provided as Figure 2. With this additional subroutine, the MAAP code was ready to be compiled and executed.

2.2 Installation of the MAAP Code on a Personal Computer

MAAP is an integrated severe accident analysis program which provides a useful tool for performing accident progression assessments such as those associated with level 2 probabilistic risk assessments. MAAP was initially developed using the Digital Equipment VAX environment and has been ported to IBM, Control Data Corporation (CDC) and other mainframe configurations. With the development of the 80386 architecture and FORTRAN compilers which can make use of expanded or extended memory, it is now possible to execute the MAAP code using a desktop computer.

In order to execute the MAAP code, however, changes to the source code are needed to address issues related to the input/output routines and file control. Also, the code must be benchmarked to results obtained for sample problems. These changes are in addition to those associated with the screen subroutine.

Table 3
Status Flags Monitored by Screen Display

Event Monitored	MAAP Status Flag
High pressure injection operation	5
Low pressure injection operation	6
Primary SRV lift	42
Primary PORV lift	44
Auxiliary feedwater flow	154
Main feedwater flow	157
Normal charging flow	11
RWST/recirculation	181
Condensate feedwater source	191
Containment spray operation	103
Core uncovery	49
Reactor vessel failure	3
Water in reactor cavity	57
Hydrogen burn (upper compartment)	102
Hydrogen burn (lower compartment)	75
Containment integrity (upper compartment)	119, 120 (depending on failure mode addressed)
Containment integrity (lower compartment)	129, 130 (depending on failure mode addressed)

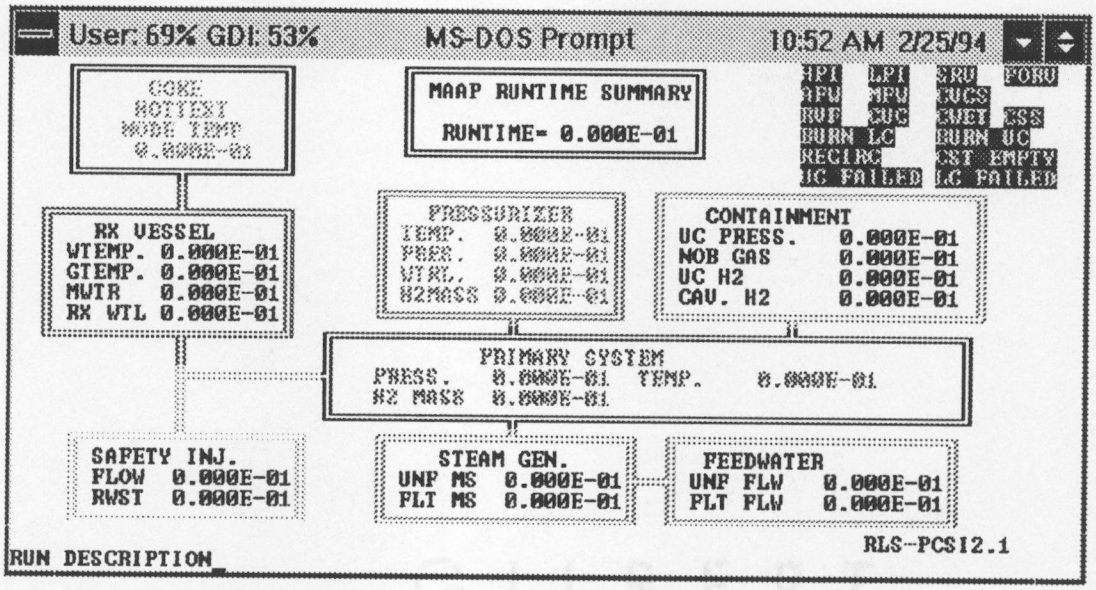

Figure 2. Sample Graphical Screen for PC Version of MAAP

The choice of FORTRAN compilers for the PC version was an important consideration in determining the number and type of changes necessary to load and execute MAAP on the PC. First, the MAAP code is too big to use standard FORTRAN compilers and required a compiler linked to an extended memory manager. Using a compiler of this type allows access to memory above the traditional 640K MS-DOS limit.

The compiler chosen, NDP FORTRAN 386,[17] provided access to the extended memory necessary for compiling the MAAP code. The compiler also contained an input/output structure similar to that present in the VAX environment. As a result, no special file control was necessary.

Some change to the basic file calls, however, was required to route the default input/output files to the appropriate location. For example, two files were not assigned in the VAX version. The variables shown below were added to the MAAP main subroutine to assign the files to appropriate unit numbers.

- ISTR = 3 (tells the computer that the sequence file is FORTRAN unit 3)
- IDIAG = 8 (sets the MAAP diagnosis file to be FORTRAN unit 8)

With the input/output defined, the next step was to compile the program. Even though the compiler provided a substantial increase in memory size, the MAAP code could not be compiled as a single source code. In order to compile the code the source code was separated into independent blocks. The MAAP code is highly structured and contains many subroutines which are controlled by a single main routine. As such, it was relatively easy to divide the MAAP code into smaller code blocks which were compiled individually and then linked into a single executable file. After compilation the executable program was then benchmarked as is discussed in Section 2.3.

2.3 Benchmark Exercise Using Standard Problem

The MAAP code requires two data files in order to execute a calculation. The first file contains the plant parameter file. This file contains 19 different groups that identify important plant parameters (e.g., number of safety injection pumps, volume of containment, etc.) that are used by the MAAP code in the accident progression. A sequence file provides specific information related to the accident sequence being assessed (e.g., large LOCA without injection). The supporting documentation provides a parameter file for a Zion-like plant design, a sample sequence file (Figure 3), and output plot files which can be used to benchmark the results obtained from the MAAP code after installation on a given system.

```
1              /BATCH UNCERTAINTY (NOT SENSITIVITY) RUN
ZION-LIKE TMLB
1
25
0
1             /MAKE TEMPORARY CHANGES TO SOME OF THESE
7,36,1.03E6        /DONT MODEL CONTMT FAILURE USING STRAIN MODEL
7,39,0.          /DONT MODEL NORMAL LEAKAGE
2,24,3           /PUT SURGE IN BKN LOOP TO GET PROPER SYMMETRY
2,13,6           /BREAK KEY FOR BKN LOOP
2,14,.00004        /BREAK AREA FROM PUMP SEAL LOCA IN BKN LOOP
2,15,7.5         /BREAK ELEVATION IN BKN LOOP
2,52,12          /BREAK KEY FOR UNBKN LOOP
2,53,.00012        /TOTAL BREAK AREA FROM PUMP SEAL LOCA'S IN UNBKN LOOPS
2,54,7.5         /BREAK ELEVATION IN UNBKN LOOPS
0,0,0            /NO MORE CHANGES
0
0.
150000.          /FINAL TIME
10000.
205            /SHUT OFF DC AND AC POWER
1
0             /NO MORE INITIATING EVENTS
8             /INTERVENE WHEN TIME (CONDITION 8)
2700            /EQUALS .75 HRS
0             /MONITOR NO OTHER INTERVENTION CONDITIONS
209            /FAIL PRIMARY SYSTEM DUE TO PUMP SEAL LOCA'S
1
0             /NO MORE ACTIONS AT THIS TIME
10
3
0             /NO MORE INTERVENTIONS
0
```

Figure 3. MAAP Sample Problem Sequence File for Zion Station Blackout

The sample sequence file was executed using the PC version of MAAP and the results compared to those provided by the MAAP documentation. If the results for the two cases are very similar, it is reasonable to conclude that the PC version is benchmarked. If a large variance in the results is found, additional changes in the MAAP source code may be necessary.

As stated earlier, the MAAP code tracks hundreds of variables during each time step. Comparison of each variable would be very time consuming and would not provide an efficient means for benchmarking the PC version. As an alternative, several of the key parameters were compared. These variables, shown in Table 4, reflect different aspects of the models within MAAP and serve as a general indication of the accuracy of the MAAP code.

Table 4
MAAP Variables Used in Benchmark Exercise

Parameter	Example Related Phenomena
Reactor Coolant System Pressure	Direct containment heating, hydrogen generation, containment source term
Reactor Vessel Water Level	Timing of vessel failure, RCS leakage rate
In-Vessel Hydrogen Generation	Hydrogen burning, core peak temperature, core relocation
Containment Pressure	Steam production, containment heat removal, pressure challenges
Concrete Ablation Depth	Debris cooling, ex-vessel hydrogen generation, presence of concrete attack

The sample accident sequence provided for comparison involves a total loss of all ac power. The major impact of the loss of all ac power is the failure of all safety equipment due to a loss of motive power. The loss of both seal injection and thermal barrier cooling to the reactor coolant pump (RCP) seals leads to pump seal failure and a loss of coolant accident (LOCA). The sample problem assumes that the LOCA occurs 45

minutes into the accident. The inability to provide makeup to the reactor coolant system (RCS) causes the RCS water level to drop and a loss of decay heat removal. The loss of RCS inventory causes the core to uncover and the fuel temperature to increase. Steam formed in the RCS passes by the fuel assemblies and, as the fuel temperature increases, hydrogen generation from zirconium-steam reaction begins. The containment pressure increases as RCS inventory is lost to the containment.

The core melts and relocates to the lower plenum before ablating the bottom of the reactor vessel and escaping to the cavity. Little water is present in the cavity since the RWST is not injected. Core concrete interaction is initiated shortly after vessel failure. The slight delay in the initiation of concrete ablation is due to the water injected into the cavity by the passive accumulators after reactor vessel failure. Containment pressure increases due to steam and gas production until failure is reached.

Figures 4 through 8 present a comparison of the results obtained from the VAX and PC versions of the MAAP code for the parameters examined. The containment pressure traces are essentially identical for the two cases. The same is true for the other parameters with the exception of the in-vessel hydrogen generation.

The timing of important events in the accident progression were also used as a measure to benchmark the PC version. Table 5 provides the timing of some key events for the benchmark and PC versions. As the table illustrates, the timing for the two cases is very similar and no major difference is found. Minor fluctuations (<2%) in the timing are due to mathematical precision, 64 bit versus 32 bit operations, and differences between the VAX and PC architecture. Given the uncertainties present in the models, these variations are not of concern.

Table 5
Timing of Important Accident Progression Events

Event Description	VAX Timing (s)	PC Timing (s)
Steam generator dryout	6716	6716
Pressurizer solid	7347	7347
Core has uncovered	8613	8668
Reactor vessel failure	13520	13638
Containment Failure	92260	94020

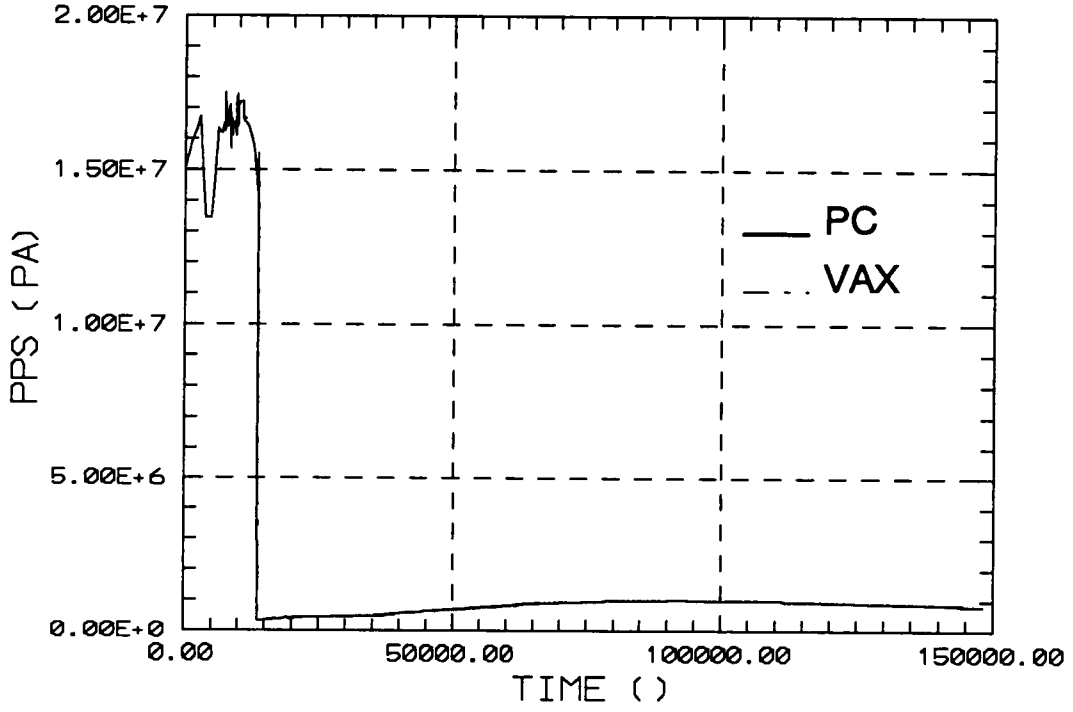

Figure 4. Comparison of RCS Pressure (PPS) for VAX and PC Calculations

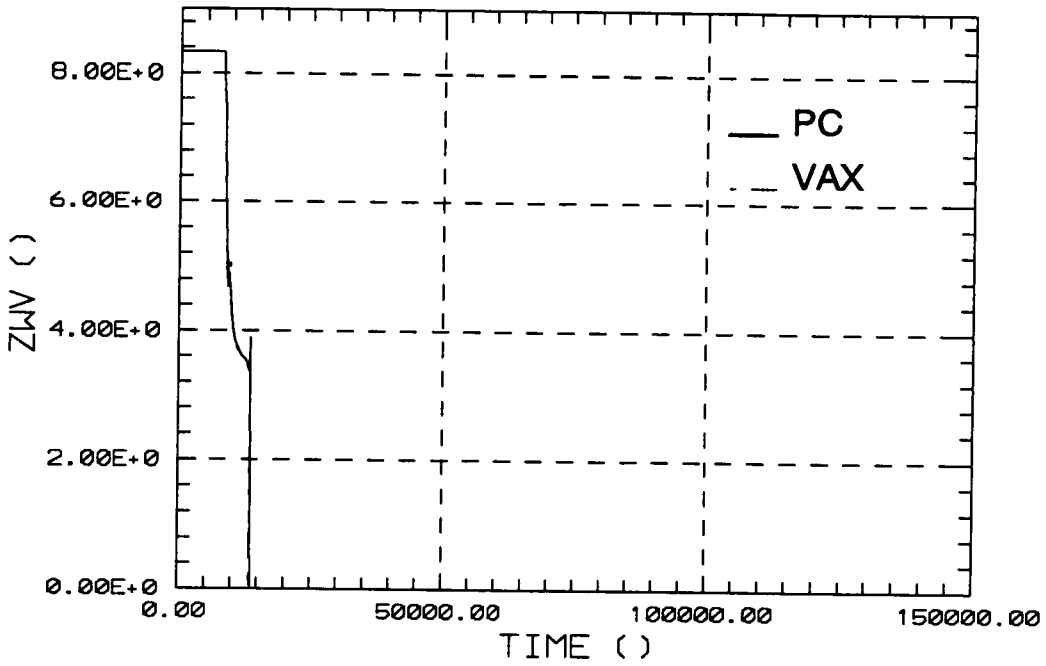

Figure 5. Comparison of Reactor Vessel Water Level (ZWV) for VAX and PC Calculations

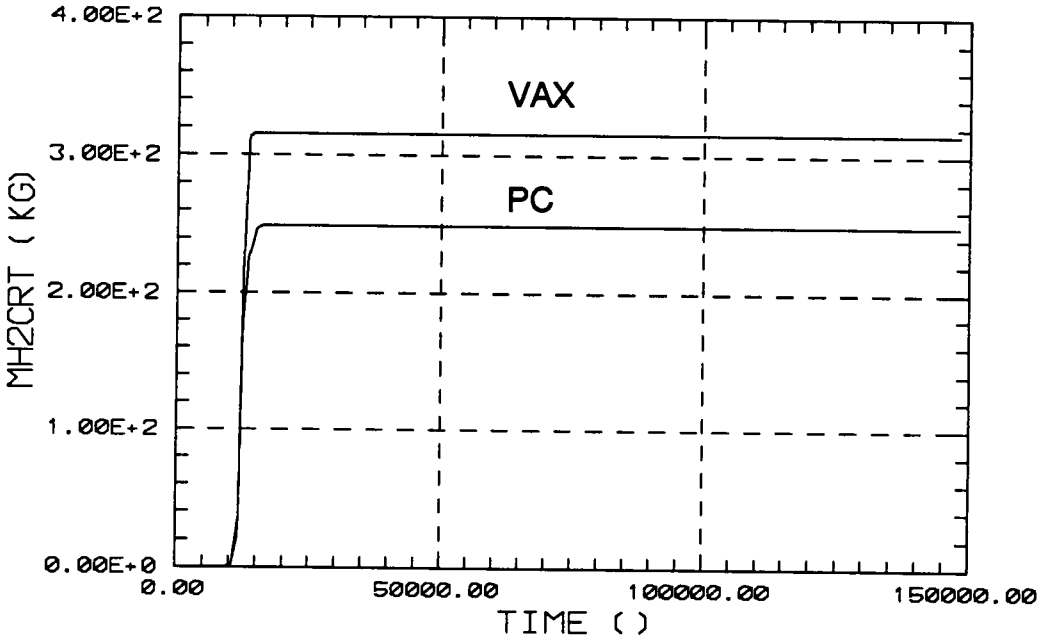

Figure 6. Comparison of In-Vessel Hydrogen Generation (MH2CRT) for VAX and PC Models

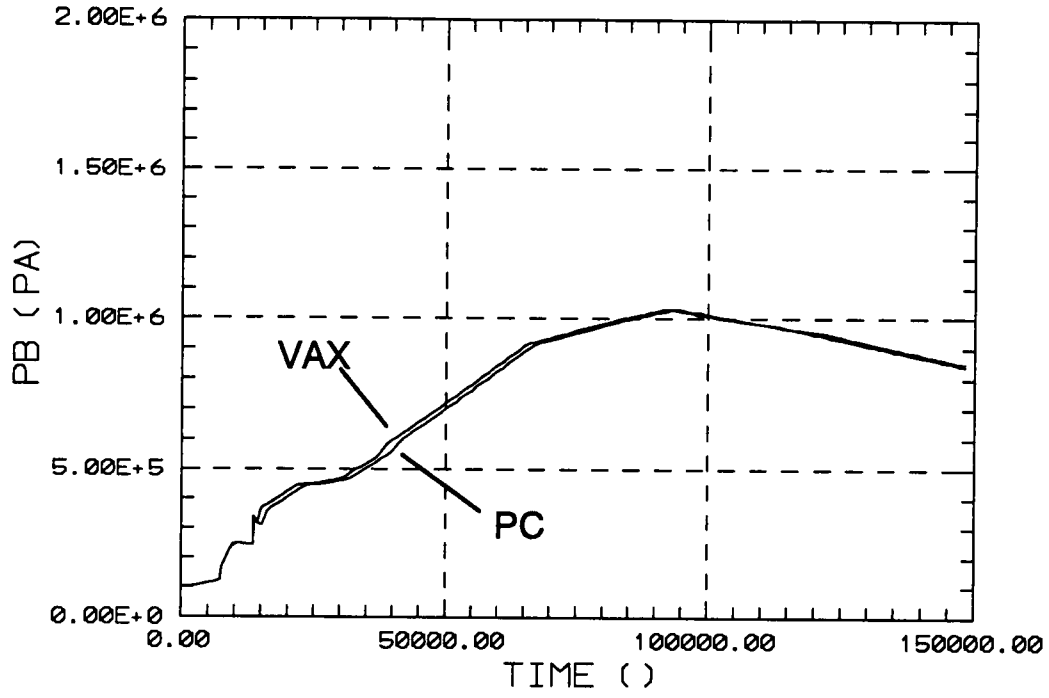

Figure 7. Comparison of Containment Pressure (PB) for VAX and PC Calculations

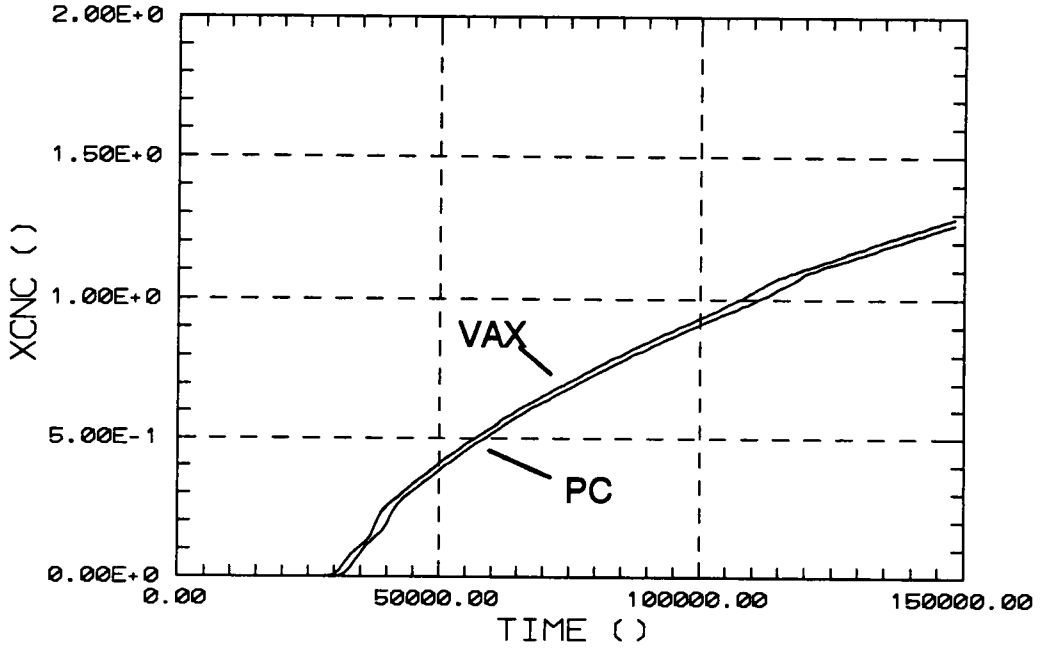

Figure 8. Comparison of Concrete Ablation (XCNC) for VAX and PC Calculations

The in-vessel hydrogen generation, Figure 6, varies substantially for the two cases. The benchmark problem generated about 70 pounds more hydrogen than the PC version and represents a 28% greater hydrogen source term. Given the importance of hydrogen in assessing the containment response to severe accidents, this magnitude of error is considered unacceptable.

An evaluation of the MAAP code routines was conducted to identify the potential cause for the difference. The conclusion drawn from this assessment is that the default time step is most likely the cause of the discrepancy.

During the final steps in the accident progression the melting debris becomes exposed to additional steam as the debris relocates to the lower support plate and finally to the lower head. This interaction can result in relatively large changes in hydrogen generation over a very short period of time. As the code steps through the time steps it is possible that the peak during a time step could be missed and a lower amount reported.

MAAP has the ability to adjust the default time step and this option was used to perform a sensitivity study on the amount of hydrogen generated in-vessel for different default time steps. Table 6 present the results of the sensitivity study. As the table shows, the amount of hydrogen generated converges to the benchmark value as the default time step is decreased. Once converged, however, additional decreases in the time step do not effect the quantity of hydrogen generated.

Table 6
Effect of Time Step on In-Vessel Hydrogen Generation

Case	Time Step (s)	In-Vessel Hydrogen Generated (lbm)
VAX	Default	315
PC-1	Default	249
PC-2	5	278
PC-3	2	307
PC-4	1	317
PC-5	0.5	317

The conclusion from the benchmark exercise is that the installation of the MAAP code onto the PC does not significantly alter expected results as long as the default time step is chosen to be 1 second or less. Time steps larger than this may lead to lower estimations of the in-vessel hydrogen source term and could impact other events which occur over a very short time period.

CHAPTER 3

SEVERE ACCIDENT ASSESSMENT OF ADVANCED PWR

3.1 Description of Approach

The approach is presented in Figure 9. The first step in the process is to load the MAAP parameter file used for the bench marking exercise. This parameter file is used to represent a current plant design. The advanced PWR design parameter file is generated by modifying the Zion-like plant parameter file to account for requirements in the Requirements Document. Prior to developing the MAAP parameter file, however, the important severe accident design features required for the advanced PWR are examined to understand the intent of information contained in the Requirements Document. The result is a plausible plant design which meets the intent of the Requirements Document.

A review of PRA studies is performed to identify the dominant accident sequences. MAAP sequence input files are then developed to simulate these accident sequences. Once complete, the accident sequences are executed for both the Zion-like design and the advanced PWR design. For the most important sequence additional cases are produced that highlight the improvement gained from various advanced PWR design features.

The results from the various cases for the Zion-like and advanced PWR models are compared on the basis of severe accident mitigation and overall containment performance. Any conclusions about the improved performance of the advanced PWR are highlighted.

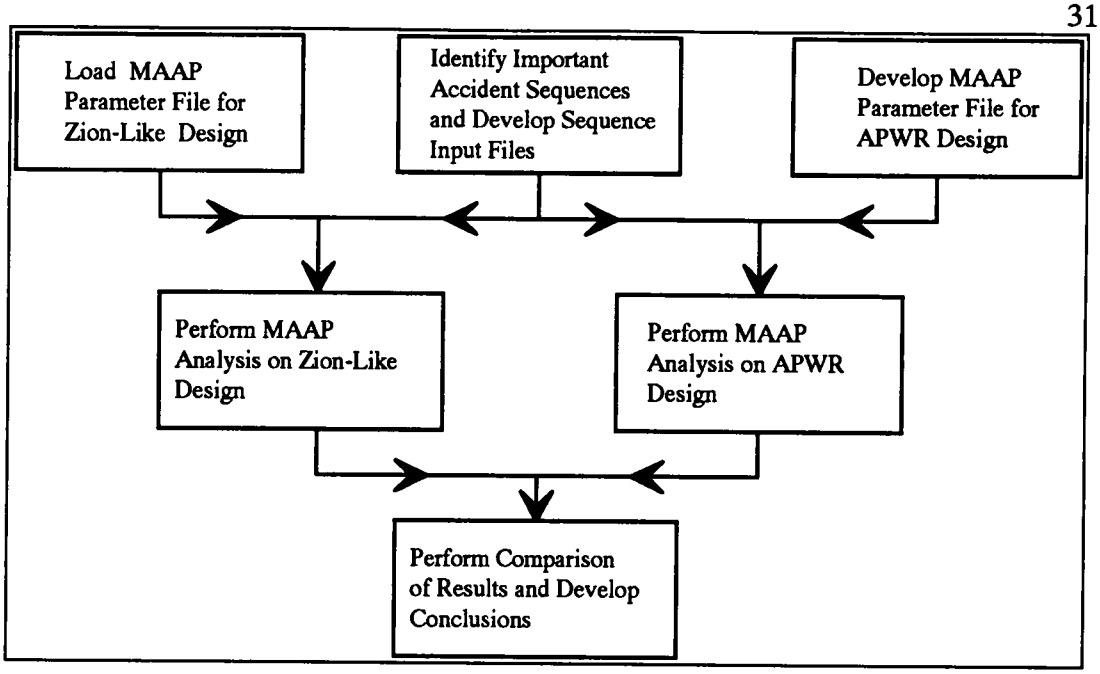

Figure 9. Comparison Process

3.2 Important Features of the Advanced PWR Design

The EPRI Requirements Document[18,19] specifies the need for several features of the advanced PWR design which are important in evaluating the design's response to a severe accident. These features include the cavity flood system, the design of the reactor cavity, and the safety depressurization and vent (SDV) system. This section provides a discussion of these features and how their implementation affects the advanced PWR response to severe accidents.

3.2.1 Cavity Flooding System

The cavity flooding system is designed to allow the operator to flood the reactor cavity either prior to or shortly after the reactor vessel has failed. Several requirements in the Requirements Document address the cavity flooding system. Although the requirements do not specify a design, they provide insights into its operation. Table 7 lists the important requirements and their interpretation for this analysis.

Table 7
Cavity Flooding System Requirements in the Requirements Document

Requirement	Interpretation
6:4.3.2.6.1[a] - Provisions shall be made to flood the lower drywell (BWR) or reactor cavity (PWR) with water to cool core debris material in the event of a severe accident.	Establishes the need for a cavity flooding system.
6:4.3.2.6.1.1 - Initial flooding of the lower drywell cavity shall be designed to be accomplished from a readily available source, in a simple way, and with sufficient water to remove decay heat, to quench the molten core debris, to account for the energy released from zirconium oxidation reaction outside the vessel, and to cover the core debris material with an overlying pool of water. For the PWR, this shall be accomplished by gravity flow from the IRWST.	This requirement establishes the IRWST as the primary means of cavity flooding and that no pumps shall be required.
6:4.3.2.6.1.2 - Flowpath(s) for delivery of water to the lower drywell (BWR) or reactor cavity (PWR) shall be designed to assure functionality for severe accidents, considering potential damage or plugging by core debris. For the PWR, a flowpath from the IRWST shall be designed for gravity flow via at least two pipes routed inside containment directly to the reactor cavity. Each pipe shall be designed for operation with a normally closed passive device or dc motor operated valve.	This requirement establishes that there are two supply lines which may contain either a passive device or a dc motor operated valve. Without additional information, a system using a dc motor operated valve is assumed. In addition, the requirement indicates that the entire system must be inside containment which prohibits manual backup actuation.
6:4.3.2.6.1.4 - Provision shall be made in the design such that initial flooding of the low drywell (BWR) or reactor cavity (PWR) shall not result in a water surface elevation high enough to wet the bottom of the reactor pressure vessel. For containment arrangements where the elevation of the source water surface is above the reactor pressure vessel lower head, the water volume initially delivered to the lower drywell or reactor cavity shall be limited to prevent wetting of the reactor vessel lower head.	This requirement does not have a significant impact on the model and is mainly included for preventing accidental RPV flooding.

Table 7 (continued)

Requirement	Interpretation
5:6.6.3.2 - The containment shall be arranged so core debris can spread over an area not less than $0.02m^2$ per megawatt rated thermal power.	Defines the cavity size such that a heat flux can be determined. The heat flux in turn is used to determine the water flow rate required to ensure debris coolability. In evaluating the size requirements for a 1300 MWe PWR, this equates to a cavity floor surface area of approximately 70 square meters (775 square ft).
6:4.3.2.6.1.4 - For long-term flooding, capability shall be provided to deliver water from the backup source outside containment (reference Chapter 5, Section 6.6.3.4) to the low drywell or reactor cavity at a rate sufficient to remove decay heat by boil-off. For the PWR, the design shall provide for adding this water volume to the IRWST.	This requirement provides for a long-term water source for the cavity flooding system. A long-term source can be used to ensure debris coolability and fission product scrubbing. This also indicates that the preferred path is through the IRWST.

Notes: a. Numerical reference to requirement in the Utility ALWR Requirements Document. Reference 6:4.3.2.6.1 refers to Chapter 6, requirement 4.3.2.6.1 of the Requirements Document.

Source: EPRI Advanced Light Water Reactor Utility Requirements Document, Chapters 5 & 6, Vol II, April 1992.

Based on the requirements, a reasonable system design incorporates two lines from the IRWST to the reactor cavity. These lines are isolated by dc-powered motor-operated valves which can be opened by the operators based on a procedural cue. The use of two independent lines allows for redundancy and is based on current standards related to safety functions. The choice of dc power for valve operation relates to the importance of total loss of all ac power sequences to the risk profile. PRAs performed on current plants during the IPE process have established that the total loss of all ac power is a dominant contributor to core damage and long-term containment failure for many operating PWRs. To improve advanced plant response following a loss of all ac power, dc power is chosen as a diverse means of valve motive power.

The cavity flooding system allows for a large quantity of water to enter the reactor cavity prior to reactor pressure vessel failure. Current severe accident research indicates that the availability of water in the cavity can provide several benefits in mitigating severe accidents which progress to reactor pressure vessel failure. The presence of water in the cavity is vital in ensuring that core debris will be coolable once the reactor pressure vessel is failed. The water acts as a cooling medium to remove decay heat from the core debris and preclude substantial interaction between the debris and the concrete. If sufficient cooling is provided, the core debris will cool and any core-concrete interaction will stop.

The interaction of water and core debris can result in rapid steam generation and the potential for steam explosions. The potential for steam explosions in the cavity sufficient to result in containment breach is considered remote and no structural failure is expected.

Interactions between the debris and the concrete result in the ablation of the concrete basemat. If concrete ablation is not controlled, the basemat will be penetrated until the debris escapes the containment and isolation is lost. As the basemat is ablated, steam and other gases are released from the concrete and increase containment pressure. Steel rebar present in the basemat can also react with the released steam in the high temperature environment and produce additional hydrogen gas which can increase the potential for hydrogen burns within the containment which increase containment pressure and possibly cause equipment to fail. Stopping the core-concrete interaction stops these processes.

Another function of the water above the core debris is to provide an effective radionuclide scrubber which will reduce ex-vessel radionuclide releases into the containment. Water has been shown to significantly reduce airborne concentrations of radionuclides by retaining them within the water. Cesium and iodine are two important radionuclides significantly reduced by water pool scrubbing.

The presence of water is not necessarily sufficient to establish debris cooling if the debris is too thick to allow efficient heat removal. It is clear that deep pools of debris are more difficult to cool than thin pools. One key factor in this observation is that the heat transfer across the debris changes dramatically between the liquid and solid state. As the debris solidifies the heat transfer across the debris is drastically reduced until a situation arises such that the water cannot remove sufficient heat to cool the core debris in the center and lower regions of the debris bed. Figure 10 presents an example of this type of configuration.

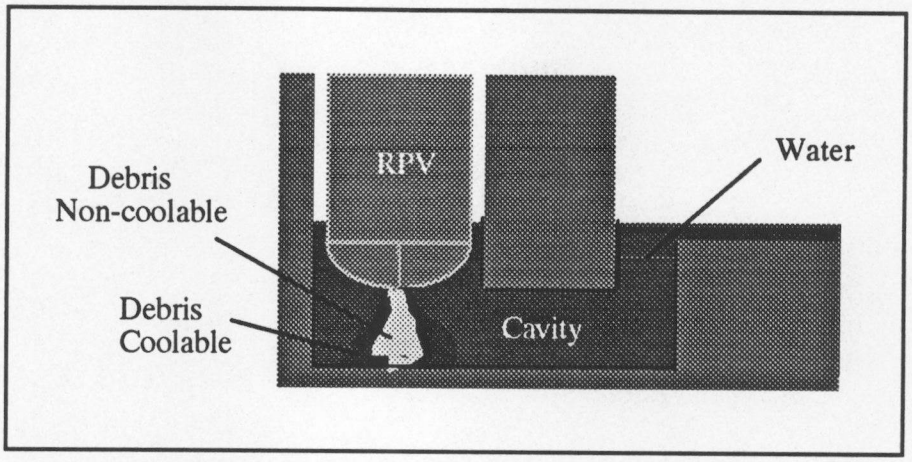

Figure 10. Potential Arrangement for Non-Coolable Debris Bed

A method to reduce the potential for deep debris beds is to spread the debris over a large space. The Requirements Document provides a cavity floor area sufficient to limit the debris thickness and improve debris cooling. Other requirements for the physical arrangement of the reactor cavity are specified and are discussed in the following section.

3.2.2 Cavity Design

Direct containment heating (DCH) has been identified as an important contributor to the potential for early containment failure for some plant designs.[20,21] DCH occurs as a result of a failure of the reactor pressure vessel while the RCS is at high pressure. Part of the core debris is entrained in the exhausting gases and swept from the cavity to the upper containment as is shown in Figure 11. The small debris particles transfer their heat to the containment resulting in pressurization. In addition, any unreacted metal can be oxidized by the steam in the containment resulting in additional hydrogen and an increased likelihood of a hydrogen burn. The peak pressure postulated for DCH with a concurrent hydrogen burn can approach the limit of containment capacity and failure may occur.

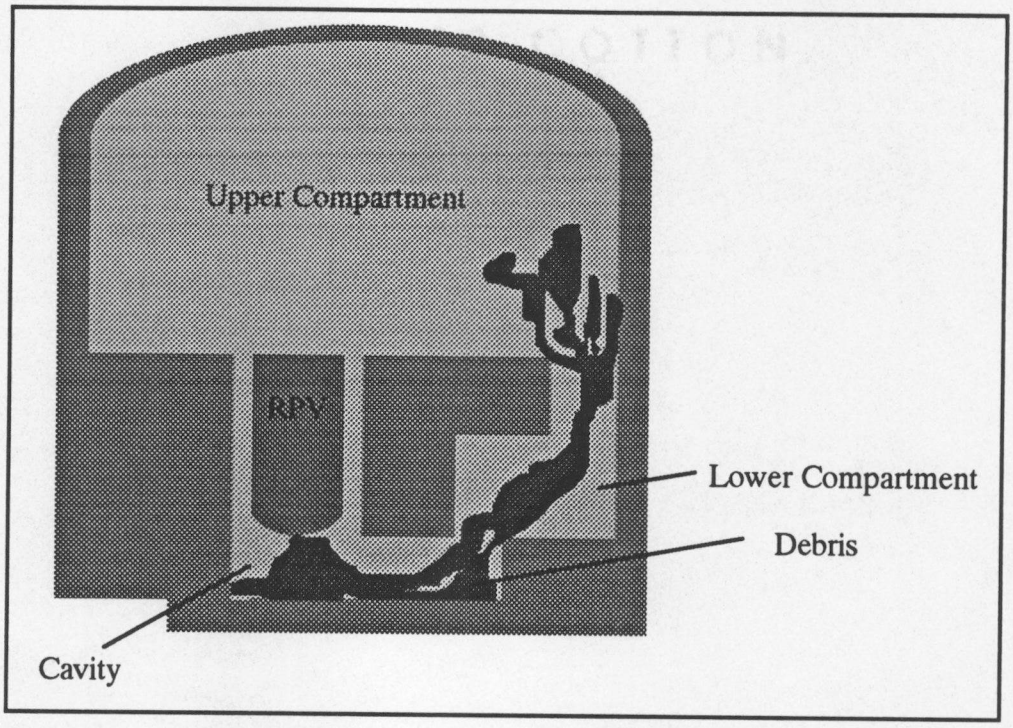

Figure 11. Debris Path from Cavity to Upper Compartment Resulting in DCH

The Requirements Document specifies a two tier approach for the minimization of the impact of DCH on the containment. The first approach is to design the reactor cavity such that the entrained debris material will be held up within the cavity. The cavity design is similar to that identified in the IDCOR program as being able to retain the highest amount of core material given a reactor pressure vessel failure at high RCS pressure.[22] The cavity design, as illustrated by Figure 12, is such that there is a right angle turn in the instrumentation chase and a collection volume is located at this point. The right angle turn should act as a separator in which the heavier core debris is separated from the exhaust gases, and the debris is trapped in the collection volume.

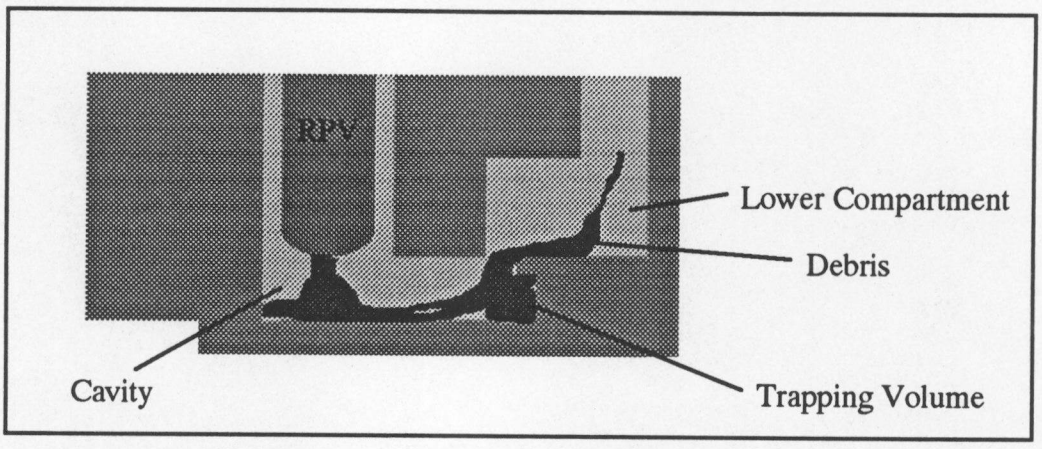

Figure 12. Example Implementation of Requirements Document Guidance on DCH

As the figure illustrates, the result is a reduction in the amount of debris which can interact with the upper compartment and a reduced pressure load. The debris trapped in the end volume would spread over the cavity floor where it could be cooled.

As discussed in the previous section, the ability to cool the core debris in the reactor cavity has a large impact on both radionuclide source term and containment integrity. The lack of definitive experimental evidence results in success of debris coolability being uncertain and an issue that is important in assessing the risk posed by an advanced PWR. The uncertainty in the ability of water to successfully penetrate and cool the core debris once the debris is located in the cavity is the subject of much work by the industry, NRC, and ARSAP.

The Requirements Document provides guidance that a system should be designed to provide sufficient water and to spread the debris over a large area such that a high confidence of debris coolability exists. The major requirement for the cavity design is that the floor surface area be such that if the core debris is covered by an overlying pool of water the thickness of the debris will not be sufficient to preclude cooling of the core debris.

3.5.3 Safety Depressurization and Vent System

In addition to the cavity design, the SDV system is provided to depressurize the RCS prior to reactor pressure vessel failure which reduces the potential for significant DCH. The SDV system is comprised of two vent paths that contain two motor-operated isolation valves and discharge to the IRWST. Each path is sufficient to provide a bleed path for feed-and-bleed cooling. The initiation of the system is specified to be manual and the valves must be capable to be opened during a station blackout. The rationale for the SDV system is discussed below.

The likelihood for DCH occurring is different for the different pressure ranges with a higher probability of significant DCH for those sequences involving high pressure. Analysis performed as a part of ARSAP[23] indicates that significant DCH can be precluded if the RCS pressure is below 750 psig. NUREG 1150[24] identified the pressure cut off for DCH as being below 200 psig and that a reduced likelihood existed for DCH for cases where the RCS pressure was between 200 and 600 psig at vessel failure.

The Requirements Document specifies several requirements on the design of the SDV system to ensure that it is capable of depressuring the RCS prior to vessel failure. Table 8 lists these requirements and their impact on the SDV design assumptions. Additionally, an ARSAP report[25] examined the size required to perform this function over the range of actuation times. The actuation of both SDV paths is required for successful depressurization. The inclusion of a means for RCS depressurization provides an additional level of protection against DCH.

Table 8
Key Requirements for the Safety Depressurization and Vent System

Requirement	Impact on the SDV System Model
5:5.5.2.3.1[a]: A single SDV system bleed path, in conjunction with two of four safety injection pumps, shall have sufficient capacity to prevent uncovering the core following a TLOFW if feed-and-bleed is initiated immediately following the opening of primary safety valves. Analyses shall show a margin of at least two feet of water above the core, using best estimate methods.	Establishes the success criterion for primary feed-and-bleed cooling.
5:5.5.2.3.1: The SDV system shall consist of two bleed paths, each capable of meeting the feed-and-bleed requirements of Section 5.5.4.3.1. The combined capacity of two SDV system bleed path divisions shall be sufficient to meet the feed-and-bleed requirements of Section 5.5.4.3.2.	This requirement specifies that there shall be two separate divisions. Each division will be capable of performing successful bleed for primary feed-and-bleed cooling.
5:5.5.4.1.2: Two valves, in series, shall be provided for each train such that vent flow can be terminated, assuming a single failure .	Defines the number of valves in each train.
5:5.5.4.1.3: The SDV system bleed valves shall be capable of operation during a station blackout.	Requires that the valves be independent of ac power. For the analysis, dc dependence is assumed.
5:5.5.4.2.1: The SDV system shall be manually actuated and no single failure shall result in inadvertent actuation.	This requires that the initiation mechanism be manual.

Notes: a. Numerical reference to requirement in the Utility ALWR Requirements Document. Reference 6:4.3.2.6.1 refers to Chapter 6, requirement 4.3.2.6.1 of the Requirements Document.

Source: EPRI Advanced Light Water Reactor Utility Requirements Document, Chapters 5 & 6, Vol II, April 1992.

3.3 Selection of Sequences for Analysis

A PRA performed on the advanced PWR design (Reference 8) is used as the basis for selecting the accident sequences for analysis. The core damage sequences are presented in Table 9 and are ranked by estimated frequency of occurrence. The three accident sequences represent 87% of the total frequency of core damage. Table 10 provides a similar listing for the Zion plant.[26]

Table 9
Advanced PWR Accident Sequence Contribution

Accident Sequences[1]	Core Damage Bin Frequency
Station blackout	3.4E-6[2]
Loss of decay heat removal	2.7E-6
Small LOCA with failure of injection	6.9E-8
Total frequency from all sources	7.1E-6

1. Dominant accident sequences taken from Reference 8.
2. Abbreviated scientific notation (1.0E-06 = 1.0×10^{-6}).

Table 10
Zion Accident Sequence Contribution

Accident Sequences[1]	Core Damage Bin Frequency
LOCAs with failure of heat removal or injection	2.7E-5[2]
Station blackout	7.9E-6
ATWS	6.6E-6
Total frequency from all sources	4.4E-5

1. Dominant accident sequences taken from Reference 26.
2. Abbreviated scientific notation (1.0E-06 = 1.0×10^{-6}).

As both tables show, station blackout sequences are very important in assessing plant performance to severe accidents. LOCAs, particularly small LOCAs, are also important. Due to their importance in both studies, these accident sequences are chosen for the plant design comparison. In addition, the loss of decay heat removal sequence is also included on the basis of its contribution to the advanced PWR core damage frequency.

3.4 Comparison of Zion-Like and Advanced PWR Features

In order to perform a comparison of the two designs, it is necessary to create a plausible advanced PWR design that can be incorporated into a MAAP parameter file. To develop this parameter file, it is necessary to understand the important design characteristics identified in the Requirements Document and how they are different than for current plants. This section discusses the major design differences that are important in defining the plant parameter file.

The Requirements Document, although providing considerable information, does not specify many of the plant specific design considerations needed to develop a detailed design model. To the extent possible, the model is developed using the requirements contained in the Requirements Document. However, in some cases specific design considerations are chosen based on current plant design practices, regulatory requirements, and engineering judgment.

Some parameters do not significantly effect the overall results of an analysis and are quite time consuming to identify. For others the design information may not be sufficient to derive a plausible value. For these cases, the Zion-like values are adopted for the advanced PWR.

The parameters contained in this file are used to define a range of characteristics such as the number of pumps in a system, the pumping curve for a pump, or the nominal

temperature of the water. The parameter file is comprised of 19 groups which are used to define the plant characteristics. Changes were necessary to 10 groups to generate an advanced PWR parameter file. Important changes can be found in seven of the ten altered groups. These changes are presented below.

The first group which was changed as a result of the advanced PWR design contains the engineered safeguards. This section defines the number of safety pumps, the pump flow curve, fan coolers, and heat exchangers. A comparison of the major equipment is provided in Table 11.

Table 11
Comparison of Engineered Safeguards

Parameter	Zion-Like Value	Advanced PWR Value
Number of safety injection pumps	3	4
Number of low head injection pumps	2	2
Number of containment fan coolers	4	0
Initial water mass in RWST (kg)	1.31E+6	4.84E+6
Number of AFW pumps	3	4
Number of containment spray pumps	1	2

The next group deals with the quench tank. The advanced PWR uses an inside containment refueling water tank. This tank also serves as the quench tank for the pressurizer depressurization valves and SRVs. As such, the initial water mass in the quench tank was increased for the advanced PWR design.

The effect of this change is that steam released from the RCS through a depressurization valve or SRV will be condensed for a much longer time than for current plants which employ a small tank.

The upper compartment group specifies the physical arrangement of the upper compartment. The upper compartment is comprised of that area of the containment between the operating deck and the top of the containment. Most of the changes identified deal with the size of the containment. In order to provide additional area for maintenance activities and refueling the containment size is increased in the APWR design. The larger free volume also contributes to severe accident mitigation by lowering the peak hydrogen concentration. The larger volume equates to a lower hydrogen concentration for a fixed hydrogen source. Hydrogen ignitors are also included in the APWR design to control hydrogen. This is a departure from present large-dry containments. The refueling pool is also larger to provide greater onsite spent fuel storage. Major parameter changes are summarized in Table 12.

Table 12
Comparison of Upper Compartment Parameters

Parameter	Zion-Like Value	Advanced PWR Value
Upper compartment free volume (ft^3)	5.89E+4	1.4E+6
Refueling pool area (ft^2)	552	1088
Height of containment spray above operating deck (ft)	43.6	91.5
Outer wall total thickness (ft)	3.0	3.0
Sedimentation area for fission product settling (ft^2)	16248	20753

Changes to the lower compartment group are similar to that found for the upper compartment group. The free volume is increased from 1.2E+4 ft^3 to 1.9E+5 ft^3. The area allowed for the debris to spread is greatly increased from 4109 ft^2 to 11466 ft^2. This increase is due to EPRI Requirements Document specifications for debris heat flux.

The cavity group (compartment C) reflects many of changes related to debris cooling. The general influence is to increase the size of the cavity to allow for better debris spreading and lower debris heat loads. The important parameter changes are highlighted in Table 13. In addition to these changes, ignitors are also present in the APWR cavity.

Table 13
Comparison of Cavity Parameters

Parameter	Zion-Like Value	Advanced PWR Value
Cavity free volume (ft^3)	7657	25000
Cavity floor area for debris spreading (ft^2)	398	1020
Height of reactor vessel above cavity floor (ft)	12.1	18.5
Flow area between cavity and lower compartment (ft^2)	63.7	49

The steam generator group identifies the improvements included in the APWR design. The major difference is an increased secondary-side volume to better accommodate transients and to provide additional time for operator response. The three major parameter changes are found in Table 14.

Table 14
Comparison of Steam Generator Parameters

Parameter	Zion-Like Value	Advanced PWR Value
Total secondary-side volume (ft^3)	5857	11330
Main feedwater temperature (F)	493	450
Nominal water level for steam generators after trip (ft)	24.6	53.67

The last group discussed is the primary system group. As with the steam generator group the key difference is an increase in the margin available. This is evident by the increased RCS volume. The volume of the APWR is about 7500 ft^3 compared to 6300 ft^3 for the Zion-like design.

3.5 Evaluation of Sequences

The sequences identified in section 3.3 are used to compare the severe accident response of the advanced PWR to that of a typical present day design represented by a Zion-like containment. MAAP sequence files are created which set the initial conditions for the MAAP code. These initial conditions include the timing of events, status of safeguards systems, and code internal parameters. Changes in the base parameter file may also be input using the sequence file. A sequence file and a plant parameter file are used to simulate the accident sequence of interest.

In the sequences examined some sequence specific considerations were made to account for certain important plant characteristics. Once the sequence file was completed, it was executed and certain parameters monitored. A comparison of these parameters is used to better understand the differences in the two designs and to draw conclusions as to improvements in the severe accident response for the APWR relative to a Zion-like design.

Specific inputs for the parameter file are highlighted in the detailed sequence description provided below along with a description of accident sequence progression.

3.5.1 Station Blackout Sequence

The total loss of all ac power sequence, commonly referred to as station blackout, is an important contributor to core damage. Since most safety system equipment requires ac power, the total loss of ac power represents a significant challenge to the coping capability of the containment. The advanced PWR has incorporated several features to improve plant performance during a total loss of ac power. These features are designed to increase the likelihood that containment will be maintained for a prolonged period of time and that the source term will be reduced.

The core melt progression for this sequence is similar for both designs. The response of the containment, however, and some safety systems is different.

Both sequences start with a loss of ac power. The loss of power means that all ac electrically-powered equipment, reactor coolant pumps, motor-driven feedwater pumps, charging pumps, etc., cease to function. The onsite batteries are used to supply dc control power until they are depleted. This allows short term operation of the auxiliary feedwater turbine-driven pump and essential plant controls.

Once dc power is lost, plant instrumentation is lost and a failure to maintain the plant in a safe-stable state is expected. The water inventory contained in the steam generators will deplete as it removes decay heat. Once the top of the steam generator tube bundle is uncovered the heat transfer rate will drop and the remaining water will slowly be steamed away.

The loss of decay heat removal results in increased temperature and pressure in the RCS which will result in a pressure relief valve challenge. This will reduce pressure and some RCS inventory will be lost through the valve. The RCS pressure will modulate as the relief valves function to maintain pressure.

Based on work performed for NUREG-1150[18], a reactor coolant pump seal LOCA will occur due to thermally-induced seal failure (assumed to occur at 45 minutes).[6] The seal LOCA will be equivalent to a small LOCA and will result in RCS depressurization and a loss of RCS inventory. The core temperature will increase and clad oxidation will commence. The rate of oxidation will be slow until the cladding temperature reaches about 1850K which is the transition point for the oxidation reaction.[27] At this point the core is uncovered and significant oxidation occurs. During core heatup the volatile radionuclides present in the fuel rods will vaporize and be released into the RCS. The radionuclides will plate out on cooler surfaces within the RCS or will escape to the containment.

RCS inventory lost through the relief valves enters the containment as steam and increases containment pressure. The loss of ac power precludes operation of containment cooling systems to lower RCS pressure. Hydrogen is also relocated to the containment through the same pathway.

The APWR design incorporates an in-containment refueling water storage tank. This tank, in addition to supplying water for flooding the refueling pit and for safety injection, serves as the discharge point for the relief valves. This large body of water, over 600,000 gallons, provides a large heat sink which condenses the steam prior to entering the containment and maintains a low containment pressure. Hydrogen gases will pass through this tank and into the containment.

Core relocation begins with the core debris relocating first to the lower support plate and then to the bottom of the reactor pressure vessel. When the lower support plate fails the debris comes into contact with water present in the bottom of the reactor pressure vessel. This interaction results in rapid steam generation which may increase RCS pressure. The debris ablates the lower head of the reactor pressure vessel and is transported to the lower compartment once the reactor pressure vessel fails. The gases present in the RCS may fragment and transport debris from the cavity to the lower compartment and other locations in the containment depending on the containment configuration. This process, called high pressure melt ejection, can result in a rapid increase in containment pressure for some containment designs.

The APWR design incorporates the SDV system which is designed to lower RCS pressure prior to reactor pressure vessel failure. The Zion-like design does not have a similar feature and the RCS pressure is maintained at the safety relief setpoint until failure.

The actuation of the SDV system is manual and actuation will occur following the occurrence of events. Present procedural guidance directs the operators to open the PORVs when the hot leg temperature exceeds 1200F. MAAP predicts that this temperature will occur on the order of 1.5 hours following the station blackout and the SDV system is assumed to actuate at this time.

Water from the accumulators will enter the cavity after the reactor pressure vessel fails and the RCS pressure is below the actuation setpoint. The water will cover the debris and promote debris cooling for some period of time. Once the water is boiled away the debris will heat up and begin to ablate the concrete containment. The reaction between the core debris and concrete releases trapped gases in the concrete and heat.

This further increases containment pressure until failure occurs due to either basemat failure or overpressure. The reaction also releases additional radionuclides which will increase the source term present in the containment.

The cavity flooding system present in the APWR design is designed to be manually actuated near the time of reactor pressure vessel failure. The purpose of this system is to add water to the cavity to promote debris cooling and to solidify the debris prior to significant concrete attack. The cavity flooding system is actuated at 9600 seconds; this is an average time predicted by MAAP at which RPV failure occurs for present designs.

Two baseline MAAP cases were executed which examined the containment response of a Zion-like plant and an advanced PWR design to a station blackout sequence. The Zion-like sequence file is the same as the benchmark run discussed in section 2.3.

The completed advanced PWR sequence file is shown in Figure 13. As illustrated, the sequence file is more complicated than the Zion-like plant file. The MAAP code allows a single event code (205) to turn off all equipment to simulate a station blackout for present plants and it is used for the Zion-like plant design.

The advanced plant design, however, cannot make use of this event code since some equipment is still available. For example, the Zion-like design PORVs fail following a loss of ac power. The SDV system performs a similar function for the advanced PWR and is dc powered. Application of the global event code in MAAP would turn the SDV system off, which is incorrect for the advanced PWR design. In fact, the SDV system is provided to allow for RCS depressurization prior to RPV failure for accident sequences such as station blackout. Another addition is the cavity flooding system. The sequence file includes the changes in the plant parameter file and the event actuation logic necessary to simulate this system.

```
1              /BATCH UNCERTAINTY (NOT SENSITIVITY) RUN
ALWR-LIKE SBO
1
25
0
1             /MAKE TEMPORARY CHANGES TO SOME OF THESE
7,36,1.03E6        /DONT MODEL CONTMT FAILURE USING STRAIN MODEL
7,39,0.          /DONT MODEL NORMAL LEAKAGE
2,24,3           /PUT SURGE IN BKN LOOP TO GET PROPER SYMMETRY
2,13,6           /BREAK KEY FOR BKN LOOP
2,14,.00004        /BREAK AREA FROM PUMP SEAL LOCA IN BKN LOOP
2,15,7.5         /BREAK ELEVATION IN BKN LOOP
2,52,12          /BREAK KEY FOR UNBKN LOOP
2,53,.00012        /TOTAL BREAK AREA FROM PUMP SEAL LOCA'S IN UNBKN LOOPS
2,54,7.5         /BREAK ELEVATION IN UNBKN LOOPS
12,03,1.0        /DECREASE MAX TIME STEP
12,05,.01          /DECREASE RELATIVE MASS CHANGE SELECT TIME STEP
6,150,3000.0       /ADD VOLUME TO CAVITY INJECTION TANK (m3)
6,151,31.6         /PROVIDE CAVITY INJECTION FLOW RATE (m3/sec)
0,0,0            /NO MORE CHANGES
0
0.
86400.           /FINAL TIME
3600.
216            /SHUT OFF HPI DUE TO NO AC POWER
1
217            /SHUT OFF LPI DUE TO NO AC POWER
1
221            /SHUT OFF CFC DUE TO NO AC POWER
1
222            /SHUT OFF CSS DUE TO NO AC POWER
1
223            /SHUT OFF PRZ SRAY DUE TO NO AC POWER
1
228            /SHUT OFF MFW DUE TO NO AC POWER
1
224            /SHUT OFF AFW DUE TO NO AC POWER
1
232            /SHUT OFF CVCS DUE TO NO AC POWER
1
0            /NO MORE INITIATING EVENTS
8            /INTERVENE WHEN TIME (CONDITION 8)
2700           /EQUALS .75 HRS
0            /MONITOR NO OTHER INTERVENTION CONDITIONS
209            /FAIL PRIMARY SYSTEM DUE TO PUMP SEAL LOCA'S
1            /SET FAILED RCS FAILED TRUE
0            /NO MORE ACTIONS AT THIS TIME
8            /INTERVENE WHEN TIME
5400           /IS AT 5400 SECONDS
0            /NO OTHER CONDITIONS
211            /AND LET OPERATORS OPEN SDS VALVES
1            /LOCK PORVS OPEN
0            /NO MORE ACTIONS
8            /INTERVENE WHEN PROBLEM TIME
9600           /IS AT 9600 Seconds (Prior to RPV failure)
0            /MONITOR NO OTHER INTERVENTION CONDITIONS
241            /CAVITY INJECTION
1            /START INJECTION
0            /NO MORE INTERVENTIONS
```

Figure 13. MAAP Station Blackout Sequence File for APWR

As discussed previously, the actuation times for the SDV and cavity flooding systems are estimated based on extrapolation of present procedural guidance and are considered reasonable. The actual procedures, when written, may specify different actuation criteria.

3.5.2 Loss of Decay Heat Removal Sequence

The loss of decay heat removal sequence is exemplified by the TMI event.[28,29] The loss of secondary-side heat removal results in a loss of decay heat removal and RCS temperature and pressure increase. If the operators do not initiate feed and bleed cooling using the PORVs and safety injection, core damage will occur. The sequence progression is very much like that for station blackout.

The major distinction is that a RCP seal LOCA does not occur and reactor pressure vessel failure occurs somewhat later in the accident. The containment response is also similar to a station blackout.

The features discussed for the advanced PWR provide similar functions as described for the station blackout sequence. The SDV system is used as the bleed path for feed and bleed cooling.

The two sequence files for the MAAP cases are presented in Figures 14 and 15. Both of these files are very similar with the APWR design including information necessary to actuate the SDV and cavity flooding systems.

```
1              /BATCH UNCERTAINTY (NOT SENSITIVITY) RUN
ZION-LIKE LOSS OF MAIN FEEDWATER AND PFB COOLING
1
25
1
1             /MAKE TEMPORARY CHANGES TO SOME OF THESE
7,36,1.03E6        /DONT MODEL CONTMT FAILURE USING STRAIN MODEL
7,39,0.          /DONT MODEL NORMAL LEAKAGE
12,03,1.0        /DECREASE MAX TIME STEP
12,05,.01          /DECREASE RELATIVE MASS CHANGE SELECT TIME STEP
0,0,0            /NO MORE CHANGES
0
0.
150000.          /FINAL TIME
3600.
205            /SHUT OFF DC AND AC POWER
1
216            /SHUT OFF HPI
1
217            /SHUT OFF LPI
1
224            /SHUT OFF AFW
1
228            /SHUT OFF MFW
1
221            /FANS FORCED OFF
1
222            /CSS FORCED OFF
1
232            /CVCS FORCED OFF
1
227            /MANUAL TRIP
1
0             /NO MORE INITIATING EVENTS
0             /NO MORE ACTIONS AT THIS TIME
0             /NO MORE INTERVENTIONS
0
```

Figure 14. MAAP Loss of Decay Heat Removal Sequence File for Zion-Like Design

```
1              /BATCH UNCERTAINTY (NOT SENSITIVITY) RUN
ALWR-LIKE LOSS OF MAIN FEEDWATER AND PFB COOLING
1
25
1
1             /MAKE TEMPORARY CHANGES TO SOME OF THESE
7,36,1.03E6          /DONT MODEL CONTMT FAILURE USING STRAIN MODEL
7,39,0.            /DONT MODEL NORMAL LEAKAGE
12,03,1.0          /DECREASE MAX TIME STEP
12,05,.01            /DECREASE RELATIVE MASS CHANGE SELECT TIME STEP
6,150,3000.0        /ADD VOLUME TO CAVITY INJECTION TANK (m3)
6,151,31.6          /PROVIDE CAVITY INJECTION FLOW RATE (m3/sec)
0,0,0              /NO MORE CHANGES
0
0.
150000.            /FINAL TIME
3600.
205              /SHUT OFF DC AND AC POWER
1
216              /SHUT OFF HPI
1
217              /SHUT OFF LPI
1
224              /SHUT OFF AFW
1
228              /SHUT OFF MFW
1
221              /FANS FORCED OFF
1
222              /CSS FORCED OFF
1
232              /CVCS FORCED OFF
1
227              /MANUAL TRIP
1
0              /NO MORE INITIATING EVENTS
8              /INTERVENE WHEN TIME
5400             /IS AT 5400 SECONDS
0              /NO OTHER CONDITIONS
211             /AND LET OPERATORS OPEN SDS VALVES
1              /LOCK PORVS OPEN
0              /NO MORE ACTIONS
8              /INTERVENE WHEN PROBLEM TIME
9600             /IS AT 9600 Seconds (Prior to RPV failure)
0              /MONITOR NO OTHER INTERVENTION CONDITIONS
241             /CAVITY INJECTION
1              /START INJECTION
0              /NO MORE INTERVENTIONS
```

Figure 15. MAAP Loss of Decay Heat Removal Sequence File for APWR

3.5.3 Small LOCA with Failure of Injection

Following a small LOCA, the safety injection system actuates to respond to the LOCA. The size of the LOCA is insufficient to lower RCS pressure sufficient to activate the low pressure safety injection system and the high pressure injection system is required. If the safety injection system fails to actuate, RCS inventory will be lost until the core is uncovered and core heatup begins. Natural circulation will be lost when the loss of RCS inventory is sufficient to void the hot legs.

With the loss of circulation within the RCS, water present in the steam generator will not be depleted. The presence of water in the steam generator will provide cooler surfaces for the radionuclides to plate out on and reduce the radionuclides present in the containment. Other than this change and somewhat different event timing, the core melt progression is similar to the other sequences.

The actuation of containment spray will occur when the containment pressure increases above the actuation setpoint. Water from the sprays will enter the cavity prior to reactor pressure vessel failure. The water will cover the debris and promote debris cooling. Water will be replenished as long as the sprays are operating and maintain debris coverage. The sprays will also condense steam and maintain a low containment pressure. The MAAP sequence files are provided in Figures 16 and 17.

```
1              /BATCH UNCERTAINTY (NOT SENSITIVITY) RUN
ZION-LIKE SMALL BREAK LOCA (2") W/O INJECTION
1
25
0
1            /MAKE TEMPORARY CHANGES TO SOME OF THESE
7,36,1.03E6        /DONT MODEL CONTMT FAILURE USING STRAIN MODEL
7,39,0.         /DONT MODEL NORMAL LEAKAGE
2,24,3          /PUT SURGE IN BKN LOOP TO GET PROPER SYMMETRY
2,13,7          /BREAK KEY FOR BKN LOOP
2,14,.004         /BREAK AREA FROM 2" LOCA IN BKN LOOP
2,15,8          /BREAK ELEVATION IN BKN LOOP
12,03,1.0       /DECREASE MAX TIME STEP
12,05,.01         /DECREASE RELATIVE MASS CHANGE SELECT TIME STEP
0,0,0          /NO MORE CHANGES
0
0.
150000.         /FINAL TIME
10000.
232           /CHARGING PUMPS
1           /FORCED OFF (FAILED)
216           /HIGH PRESSURE INJECTION
1           /FORCED OFF (FAILED)
217           /LPI PRESSURE INJECTION
1           /FORCED OFF (FAILED)
224           /AUXILIARY FEEDWATER
1           /FORCED OFF (FAILED)
209           /PIPE BREAK
1           /INITIATE
0           /NO MORE INITIATING EVENTS
```

Figure 16. MAAP Small LOCA Sequence File with Failure of Injection for the Zion-Like Design

```
1              /BATCH UNCERTAINTY (NOT SENSITIVITY) RUN
ALWR-LIKE SMALL BREAK LOCA W/O INJECTION
1
25
0
1             /MAKE TEMPORARY CHANGES TO SOME OF THESE
7,36,1.03E6          /DONT MODEL CONTMT FAILURE USING STRAIN MODEL
7,39,0.          /DONT MODEL NORMAL LEAKAGE
2,24,3           /PUT SURGE IN BKN LOOP TO GET PROPER SYMMETRY
2,13,7           /BREAK KEY FOR BKN LOOP
2,14,.01         /BREAK AREA FOR LOCA IN BKN LOOP
2,15,8           /BREAK ELEVATION IN BKN LOOP
12,03,1.0        /DECREASE MAX TIME STEP
12,05,.004         /DECREASE RELATIVE MASS CHANGE SELECT TIME STEP
6,150,3000.0     /ADD VOLUME TO CAVITY INJECTION TANK (m3)
6,151,31.6       /PROVIDE CAVITY INJECTION FLOW RATE (m3/sec)
0,0,0            /NO MORE CHANGES
0
0.
86400.           /FINAL TIME
3600.
232             /CHARGING FLOW
1              /FORCED OFF (FAILED)
216             /HIGH PRESSURE INJECTION
1              /FORCED OFF (FAILED)
217             /LPI PRESSURE INJECTION
1              /FORCED OFF (FAILED)
224             /AUXILIARY FEEDWATER
1              /FORCED OFF (FAILED)
209             /INITIATE BREAK
1
0              /NO MORE INITIATING EVENTS
8              /INTERVENE WHEN TIME
5400            /IS AT 5400 SECONDS
0              /NO OTHER CONDITIONS
211             /AND LET OPERATORS OPEN SDS VALVES
1              /LOCK PORVS OPEN
0              /NO MORE ACTIONS
8              /INTERVENE WHEN PROBLEM TIME
9600            /IS AT 9600 Seconds (Prior to RPV failure)
0              /MONITOR NO OTHER INTERVENTION CONDITIONS
241             /CAVITY INJECTION
1              /START INJECTION
0              /NO MORE INTERVENTIONS
```

Figure 17. MAAP Small LOCA Sequence File with Failure of Injection for the APWR Design

CHAPTER 4

RESULTS OF THE ANALYSIS

This section describes the results of the accident sequence analysis and documents the insights gained regarding the performance of a containment design that meets the EPRI Requirements Document.

A comparison of the results for the Zion-like design and the APWR design is provided for each of the three sequences examined. From this comparison insights are gained into how the new features of the APWR design will influence severe accident response. The comparison is divided into two categories, core damage and containment response. Several plant parameters are compared and the differences are related to unique APWR features. Table 15 lists the parameters examined.

Table 15
Plant Parameters Used for Comparison

MAAP Parameter	Description	Plot file	Pointer
MH2CRT	Hydrogen generated in-vessel	31	3
TWCR	Water temperature in the core	31	6
ZWV	Water level in the reactor vessel	31	7
TGPS	Average temperature of gas in RCS	31	13
TCRHOT	Hottest core node	31	23
PPZ	Pressure in the pressurizer	31	29
MCMC	Mass of corium in cavity	33	9
XCNC	Concrete ablation depth in cavity	33	10
PB	Pressure in lower compartment	33	12
ZWB	Water level in the lower compartment	33	15
XCNB	Concrete ablation depth in lower compartment	33	19
MH2A	Mass of hydrogen present in upper compartment	33	22
MH2B	Mass of hydrogen present in lower compartment	33	17
MH2C	Mass of hydrogen present in cavity	33	8
TCMC	Average temperature of corium in the cavity	33	3
MWC	Mass of water in the cavity	33	5

To further examine the new systems associated with the APWR design, a set of sensitivity studies is performed. These studies examine the incremental benefit of the cavity flooding and SDV systems in relation to improved severe accident response.

4.1 Station Blackout Sequence

The station blackout sequence is an important sequence for evaluating the plant severe accident response. The early failure of both core cooling and containment safeguards represents a strong challenge to the plant's ability to maintain containment integrity.

One consideration in assessing plant response is the quantity of hydrogen generated during core heatup. Hydrogen control is very important in precluding hydrogen burning. Hydrogen burns release energy in the containment and increase containment pressure. If sufficient hydrogen is present, hydrogen detonations may occur and produce dynamic loads that can overcome the containment capacity.

Figure 18 provides a summary of in-vessel hydrogen production for the station blackout cases. Hydrogen production is initiated earlier for the APWR case because the SDV operation causes core damage to occur earlier. The peak hydrogen production, however, is less for the APWR. As the RCS inventory is lost through the SDV, less steam is available in the RCS to react with the cladding and less in-vessel hydrogen is produced.

Figure 19 presents the collapsed water level in the reactor pressure vessel for the two cases. The water level for the APWR maintains a constant value until SDV activation. An almost constant depletion occurs until the RCS pressure is below the accumulator setpoint. The injection of the accumulator inventory into the RCS offsets the SDV bleed path until the accumulators are depleted and the water level again decreases.

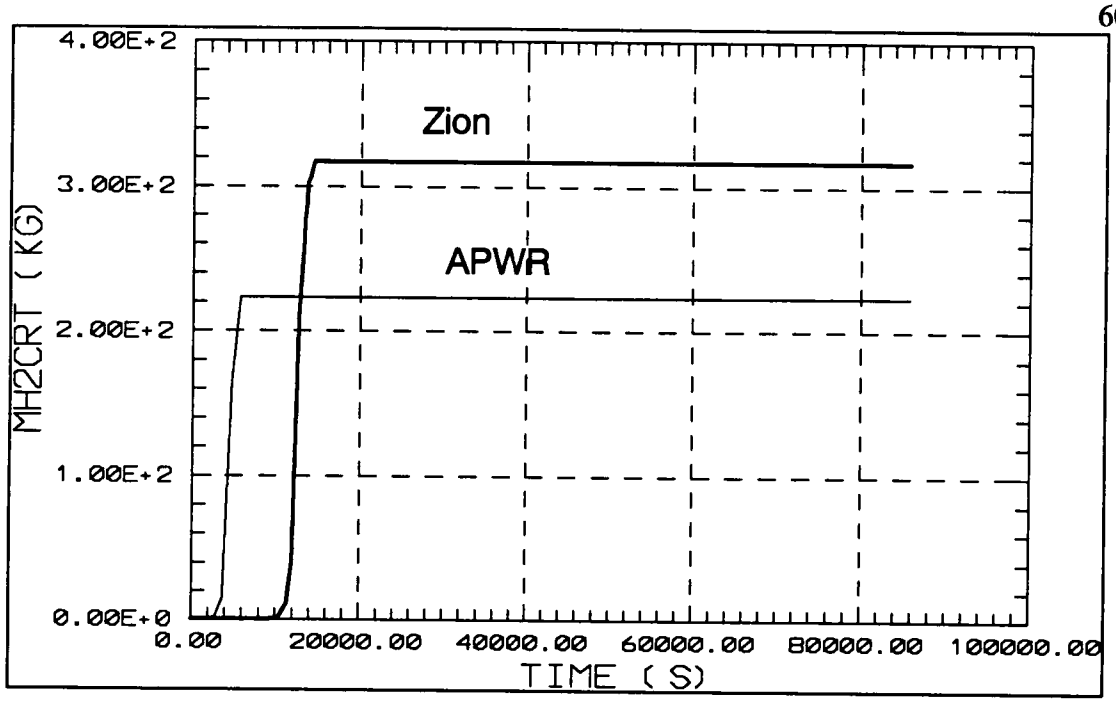

Figure 18. Hydrogen Mass Generated In-Vessel (MH2CRT) for Station Blackout Sequence

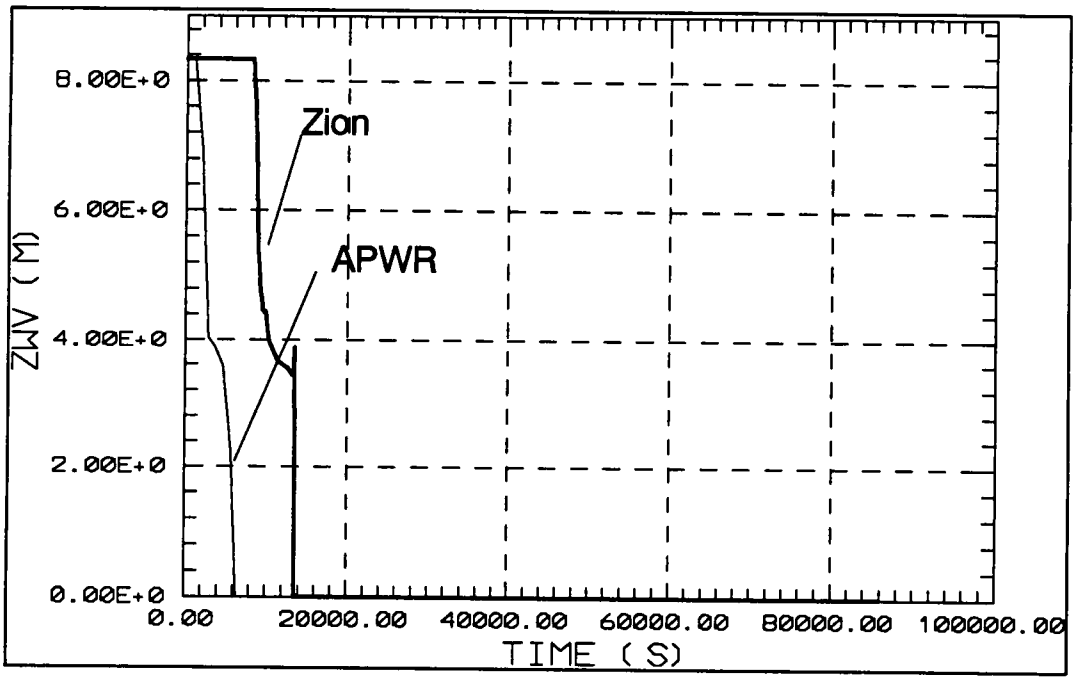

Figure 19. Collapsed Water Level in Reactor Pressure Vessel (ZWV) for Station Blackout Sequence

For the Zion-like design, the water level remains constant until the relief valves have voided the pressurizer and hot legs. The operation of the relief valves slowly depletes the water in the RCS. The sudden drop in water level is due to the failure of the reactor pressure vessel.

Although separated by time, the general shape of peak clad temperature is quite similar as shown by Figure 20. The peak temperature reached by the APWR design is slightly higher due to the lack of RCS inventory and higher decay heat load. The initial drop in peak temperature represents the relocation of melted material from the fuel matrix. The final drop occurs when the reactor pressure vessel is failed.

The RCS pressure for the Zion-like design initially drops until the steam generators are dry and decay heat removal is lost (Figure 21). RCS heatup begins and the RCS pressure increases to the safety relief valve setpoint. The valves oscillate and maintain RCS pressure until reactor pressure vessel failure occurs. The RCS pressure for the APWR design slowly decreases until decay heat removal is lost. The SDV system is actuated at this point and RCS pressure rapidly decreases to low pressure prior to reactor pressure vessel failure.

In examining the containment response it is important to consider the temperature of the debris that exits the reactor pressure vessel and enters the cavity. Figure 22 presents the average temperature of the debris in the cavity. A comparison of the early temperatures provide insight into the effectiveness of the cavity flooding system. The Zion-like design experiences a high temperature as the debris enters the essentially dry cavity.

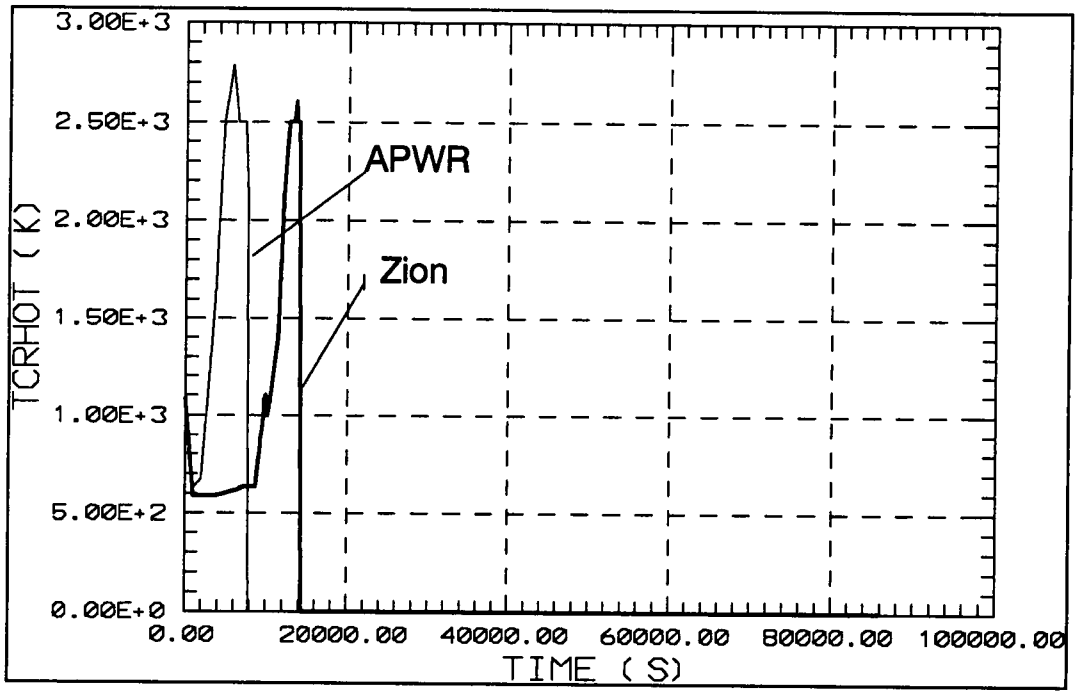

Figure 20. Peak Clad Temperature (TCRHOT) for Station Blackout Sequence (Note: the temperature is recorded as zero after reactor pressure vessel failure)

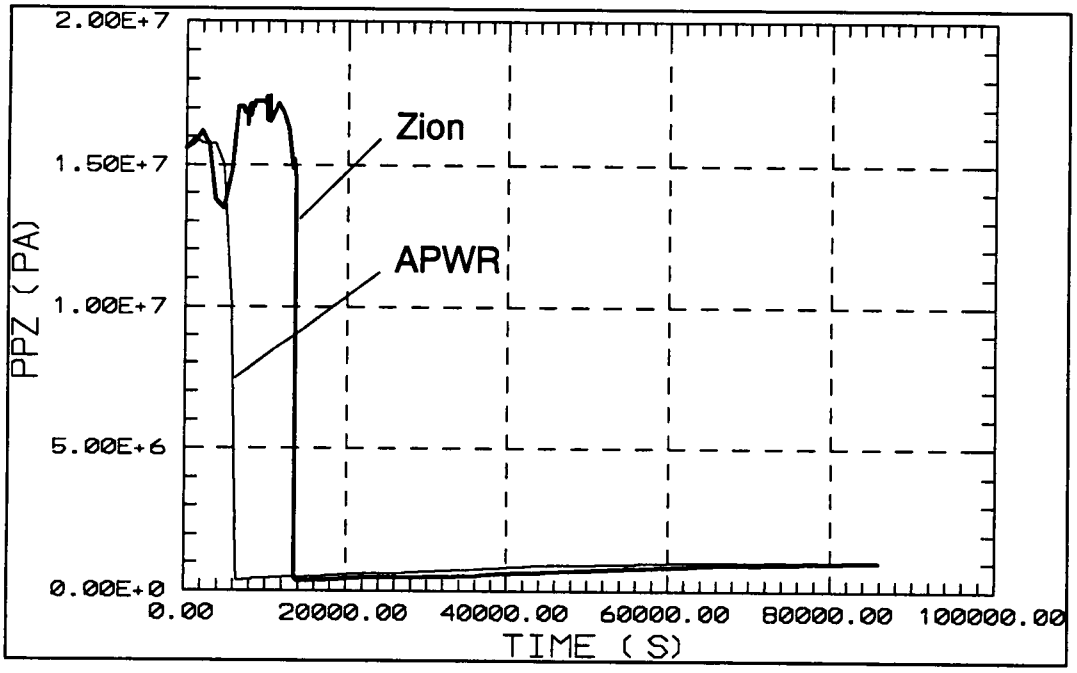

Figure 21. RCS Pressure (PPZ) for Station Blackout Sequence

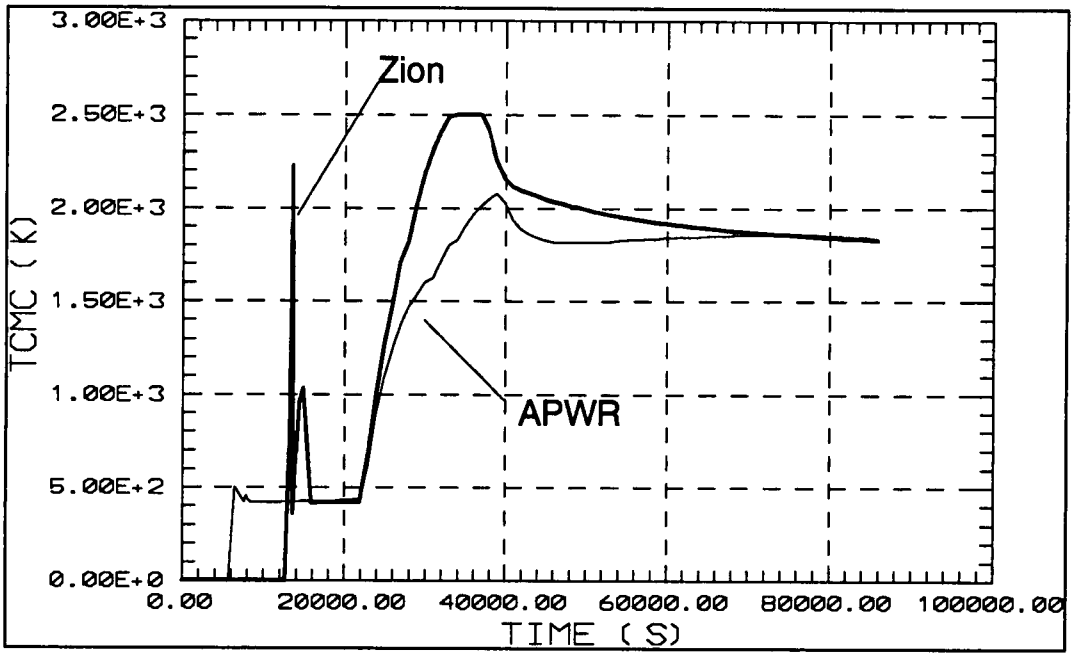

Figure 22. Temperature of Debris in Cavity (TCMC) for Station Blackout Sequence (Note: the temperature is recorded as zero prior to debris entering the cavity)

The water remaining in the RCS and the accumulators enters the cavity and cools the debris until it is steamed away. The debris temperature increases after the water is gone and reaches the melting temperature. The debris slightly cools as it interacts with the concrete in the cavity.

The APWR design is notably different with the debris temperature remaining low for the early phase of the event. This is due to the water present in the cavity that is injected by the cavity flooding system (Figure 23). The water in the cavity is boiled away and the debris temperature increases. Since the debris is at a lower decay heat level and has been cooled for a prolonged period of time, the peak temperature is lower than exhibited for the Zion-like design. The lower temperature corresponds to a slower concrete ablation rate (Figure 24) and lower hydrogen production.

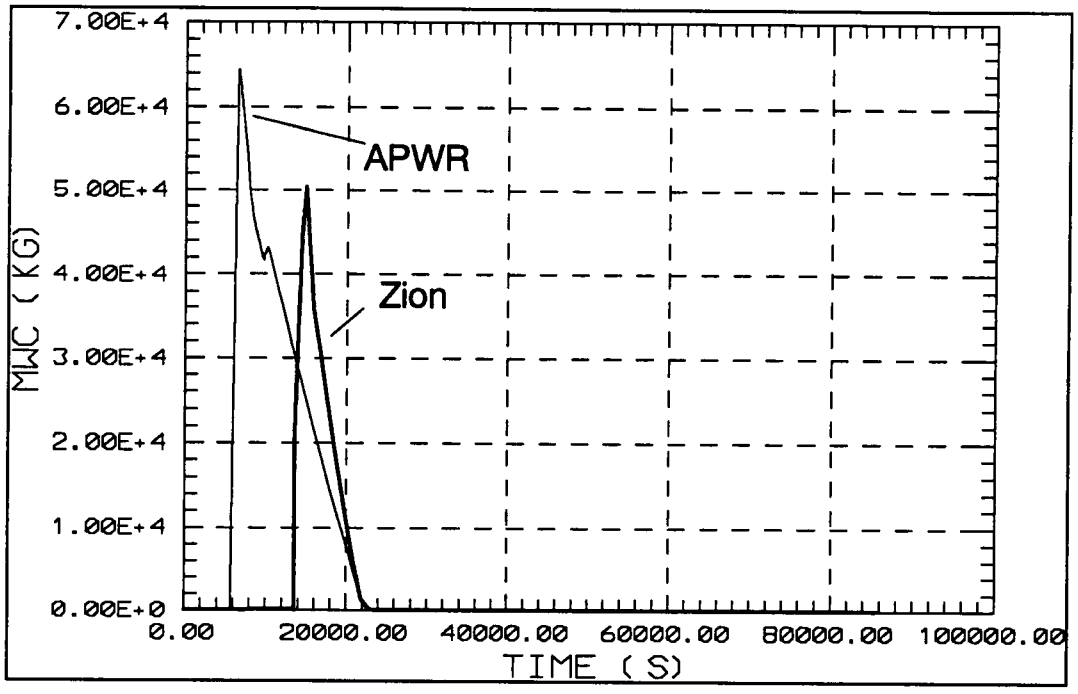

Figure 23. Water Mass in Cavity (MWC) for Station Blackout Sequence

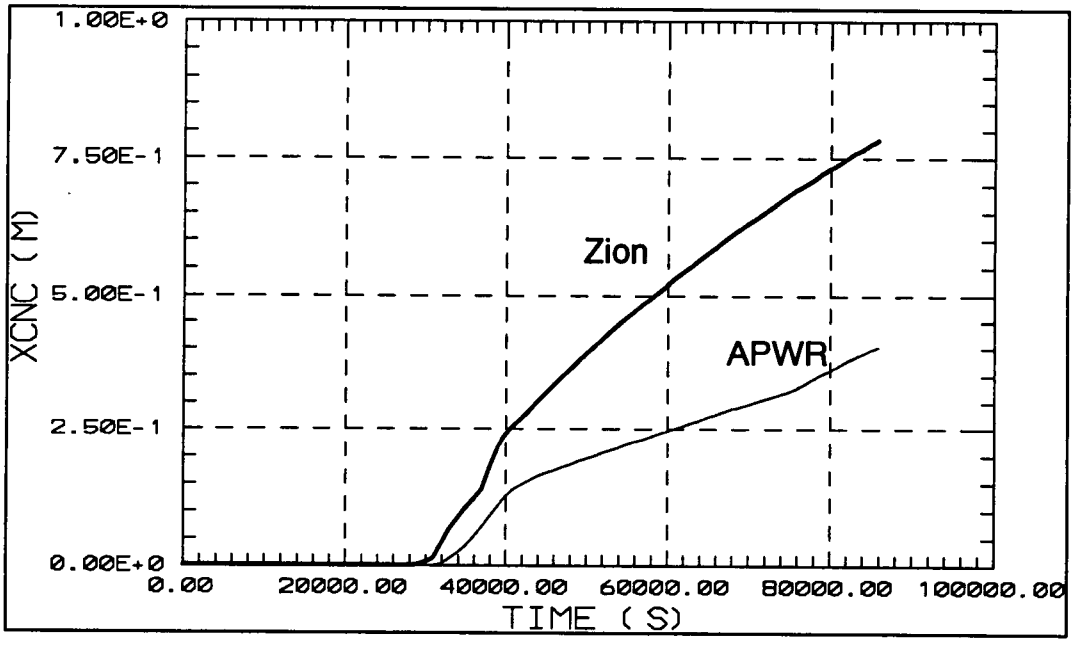

Figure 24. Concrete Ablation Depth (XCNC) for Station Blackout Sequence

The hydrogen mass in the upper containment, illustrated in Figure 25, is less for the APWR design than for the Zion-like design due to two factors. First, the in-vessel hydrogen production is less. After reactor pressure vessel failure, hydrogen production is limited by ex-vessel production mechanisms, i.e., concrete attack. The presence of water in the debris cools the debris and precludes concrete attack. As discussed earlier, concrete attack for the APWR will occur at a lower rate. Hydrogen production is caused by the release of gases in the concrete that interact with unoxidized materials in the debris and rebar in the concrete structure. The rate of production is directly related to the ablation rate. Thus, the hydrogen production rate will be reduced by a similar ratio as the ratio of concrete ablation rates. The overall result is a reduction of almost 100 percent in hydrogen generated over the period examined. This reduction substantially reduces the potential for hydrogen concentrations necessary for hydrogen burning.

The last parameter examined is the containment pressure. Figure 26 presents the pressure trace for the two containment designs. The containment pressure for the APWR design begins to increase once the SDVs are actuated. A step increase occurs in pressure when the debris interacts with the water in the cavity. The pressure then steadily increases as the water is steamed away from the cavity. Once the debris is dry, the rate of pressurization decreases to account for the change in the pressurization source. The later pressurization is due to energy and gases released during concrete attack and results in a slower pressurization rate. The Zion-like design pressure remains low until the relief valves begin to cycle. At reactor pressure vessel failure, a step increase is shown.

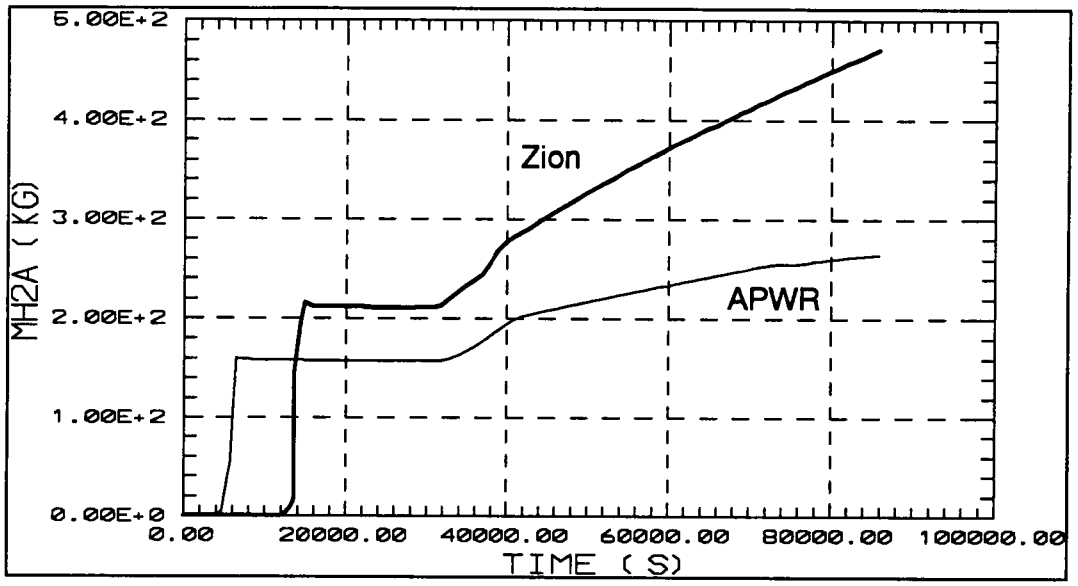

Figure 25. Hydrogen Mass in Upper Compartment (MH2A) for Station Blackout Sequence

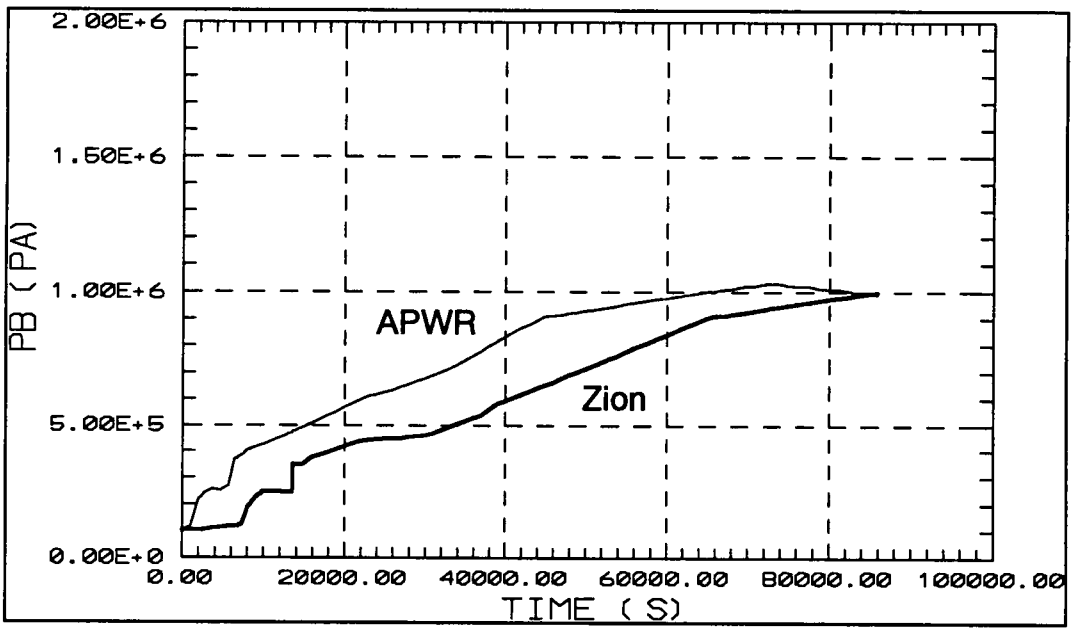

Figure 26. Representative Containment Pressure (PB) for Station Blackout Sequence

The rate of pressurization is similar to the APWR until the water is boiled off of the debris. At that point, the concrete attack provides the source of gases and energy for pressurization. As described earlier, the rate of pressurization is higher for the Zion-like design and the pressure at the end of the analysis is essentially the same by the end of the analysis.

Several insights can be obtained based on the station blackout sequence results. For the core damage phase of the event the initiation of the SDV system results in the RCS pressure being lower at the time of reactor pressure vessel failure. This is advantageous in precluding the effects of high pressure melt ejection. The removal of RCS inventory also reduces the steam available for clad oxidation and reduces the total quantity of hydrogen produced in-vessel. An adverse effect of actuating the SDV system is that reactor pressure vessel failure occurs earlier in the sequence. This provides less time for accident mitigation and recovery prior to reactor pressure vessel failure and reduces the possibility of arresting the event prior to substantial containment releases. Delaying the actuation of the SDV system can offset this effect at the cost of increased hydrogen production and a potentially higher RCS pressure at the time of reactor pressure vessel failure.

The containment analysis indicates that the operation of the cavity flooding system provides benefits by prolonging the time until concrete attack and reducing ex-vessel hydrogen production. Both of these benefits delay the time to containment failure and improve the chance for mitigation of the accident. The additional water provided over the debris in the cavity does increase containment pressure early in the event. This is offset, however, later in the sequence by reduced gas production from concrete attack such that the long term effect is small.

4.2 Loss of Decay Heat Removal Sequence

The loss of decay heat removal sequence is very similar to the station blackout sequence with the exception that an RCP seal LOCA does not occur. The failure of decay heat removal results in an early failure of the reactor pressure vessel. For the Zion-like design the failure is accompanied by high RCS pressure and high pressure melt ejection. The activation of the APWR SDV system depressurizes the RCS and precludes high pressure melt ejection.

The attributes examined for the station blackout sequence are used in evaluating the loss of decay heat removal sequence. Figure 27 provides a summary of in-vessel hydrogen production. Hydrogen production is initiated earlier for the APWR case because the SDV operation causes core heatup to occur earlier. The earlier heatup produces temperatures needed for oxidation earlier in the sequence. The APWR production ceases at about 5000 seconds due to accumulator discharge and a reflood of the reactor pressure vessel. After this water is lost, hydrogen production resumes until reactor pressure vessel failure. The peak hydrogen production is less for the APWR than the Zion-like design. The combination of reflood and less steam later in the event due to SDV actuation limits the in-vessel hydrogen produced for the APWR design.

Figure 28 presents the collapsed water level in the reactor pressure vessel. The water level for the APWR maintains a constant value until SDV activation. An almost constant depletion occurs until the RCS pressure is below the accumulator setpoint. The injection of the accumulator inventory into the RCS refloods the reactor pressure vessel. The continued actuation of the SDV system depletes the RCS inventory until failure occurs.

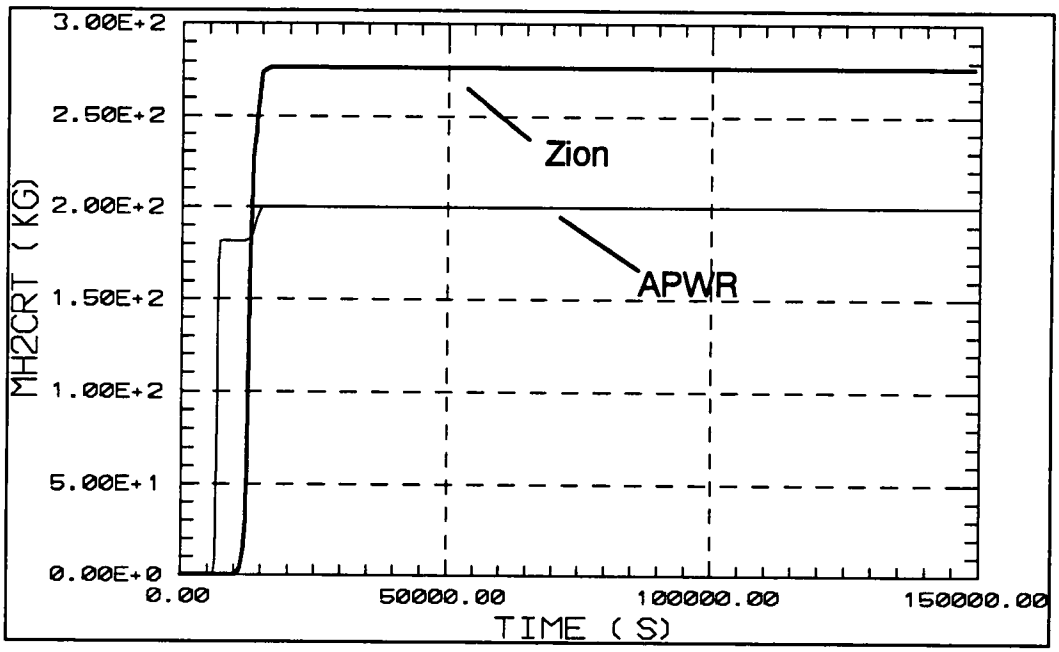

Figure 27. In-Vessel Hydrogen Production (MH2CRT) for Loss of Decay Heat Removal Sequence

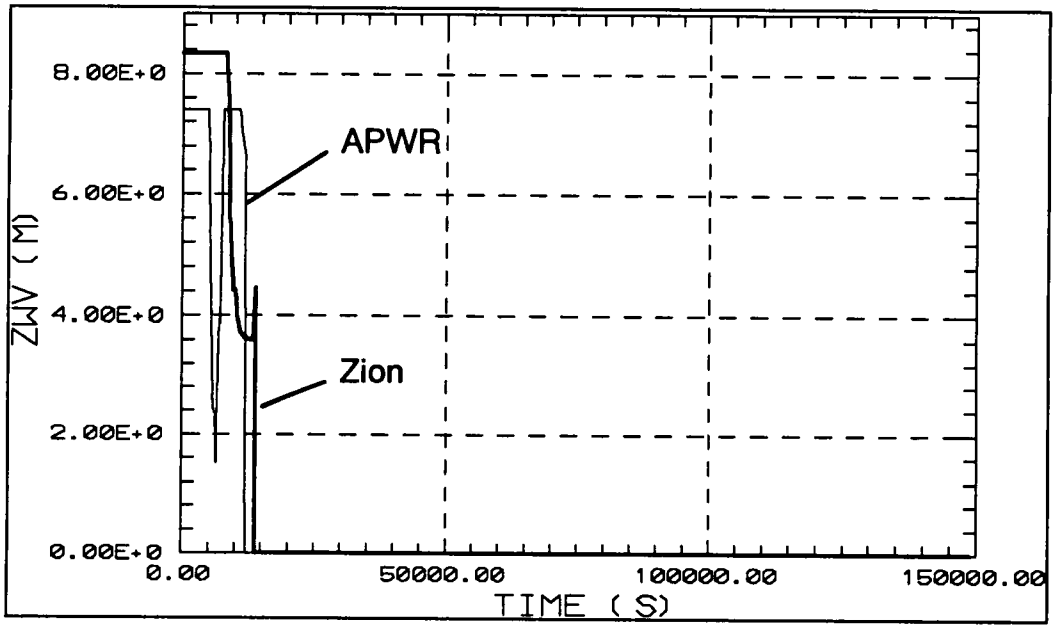

Figure 28. Collapsed RCS Water Level (ZWV) for Loss of Decay Heat Removal Sequence

For the Zion-like design, the water level remains constant until the relief valves have voided the pressurizer and hot legs. The operation of the relief valves slowly depletes the water in the RCS. A spike occurs at the point of reactor pressure vessel failure due to the actuation of the accumulators. The accumulators, however, cannot restore the coolant level due to the failure of the reactor pressure vessel.

The peak temperature, see Figure 29, reached by the APWR design is higher than the Zion-like design. The higher temperature is due to the early voiding of the RCS by the actuation of the SDV system. The decay heat load is higher for the APWR case and a higher peak temperature is experienced prior core relocation. As the accumulator inventory is boiled off, the fuel temperature decreases until the core is no longer covered by water and core heatup is resumed. The final temperatures before reactor pressure vessel failure are the same for the two cases.

The RCS pressure (Figure 30) for the Zion-like design increases as RCS heatup begins and reaches the safety relief valve setpoint. The valves oscillate and maintain RCS pressure until reactor pressure vessel failure occurs. The RCS pressure for the APWR design rapidly decreases after the actuation of the SDV system until the accumulators are actuated. The injection of water on the hot core results in steam generation and a pressure spike occurs. The RCS pressure then quickly decreases to low pressure prior to reactor pressure vessel failure.

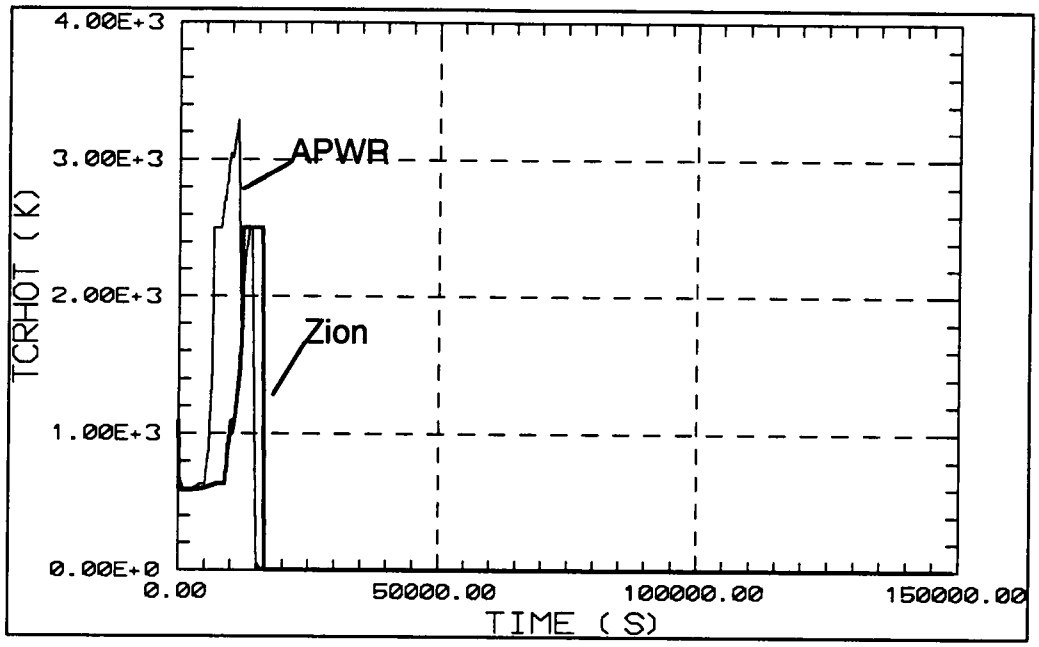

Figure 29. Peak Cladding Temperature (TCRHOT) for Loss of Decay Heat Removal Sequence (Note: the temperature is recorded as zero after reactor pressure vessel failure)

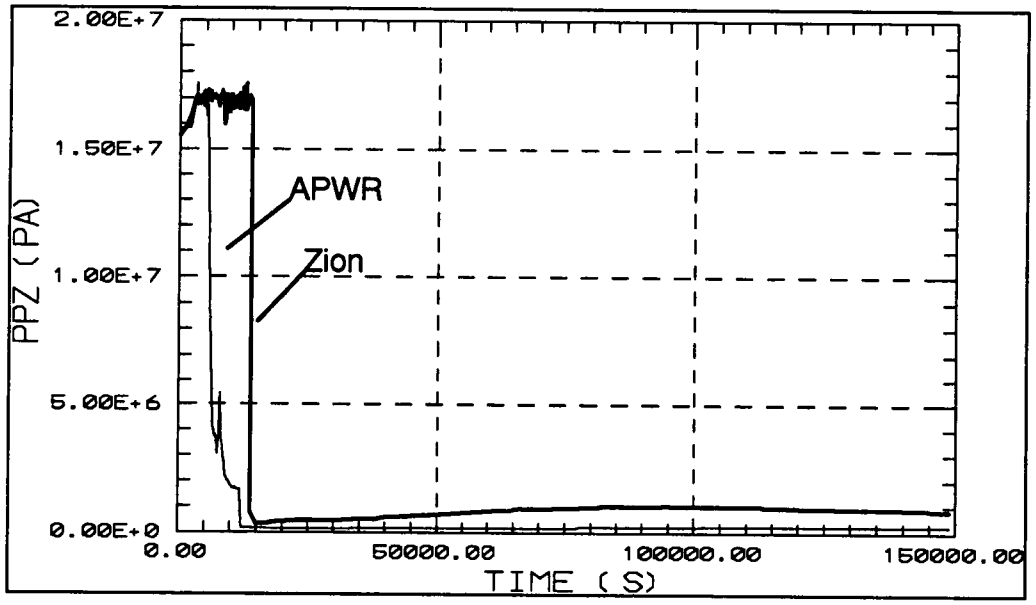

Figure 30. RCS Pressure (PPZ) for Loss of Decay Heat Removal Sequence

Following reactor pressure vessel failure the emphasis transfers from within the RCS to the containment cavity. Figure 31 presents the average temperature of the debris in the cavity. A comparison of the early temperatures provides insight into the effectiveness of the SDV system in mitigating high pressure melt ejection concerns. As the debris relocates from the reactor pressure vessel to the cavity both designs experience high temperature debris. For the APWR design, water from the cavity flooding system quenches the debris and the debris remains cool until the water is boiled off. The Zion-like design experiences a high temperature pulse which is followed by a second peak temperature. This second peak provides insights into how the SDV system reduces the potential for significant HPME and DCH.

Figure 32 presents the total debris mass present in the cavity for the Zion-like design. The figure shows an initial pulse followed by an increasing mass. The cause of this is the displacement of debris from the cavity to the lower compartment followed by relocation back to the cavity. The high pressure gases sweep the debris from the cavity to the lower compartment where it can interact with the containment and transfer energy.

Figure 33 presents the mass present in the lower compartment. Comparing Figures 32 and 33 it is clear that the gases transfer the majority of the debris to the lower compartment. After the gases pass, some of the debris present in the lower compartment relocates, by gravity, to the cavity and a corresponding increase in mass and temperature are seen.

In the case of the APWR design, the reactor pressure vessel fails at low pressure that is insufficient to entrain debris and the debris remains in the cavity. The debris relocates to the cavity and the energy released to the containment is involved in boiling the water present in the cavity and not in containment heatup.

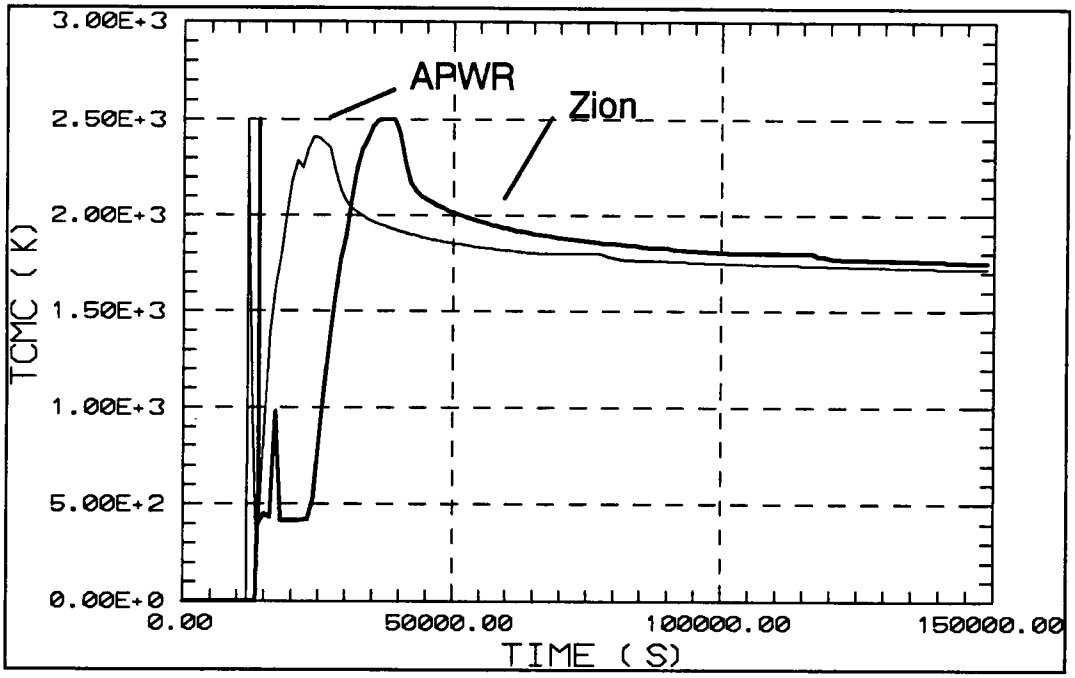

Figure 31. Debris Temperature in the Cavity (TCMC) for Loss of Decay Heat Removal Sequence (Note: the temperature is recorded as zero prior to debris entering the cavity)

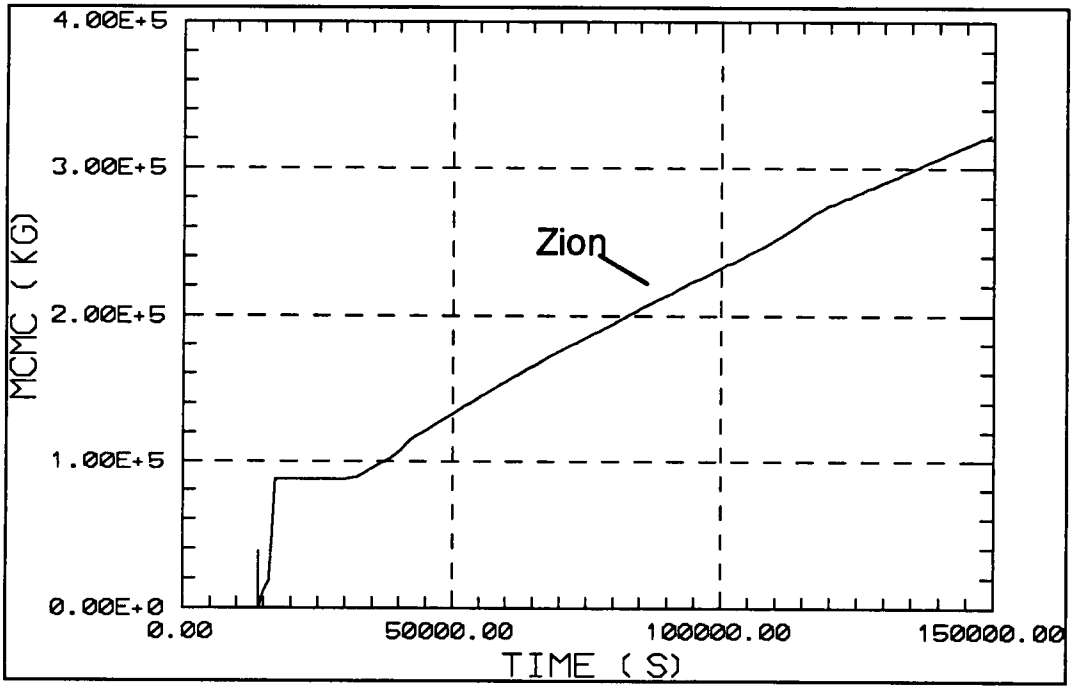

Figure 32. Debris Mass in the Cavity (MCMC) for Loss of Decay Heat Removal Sequence

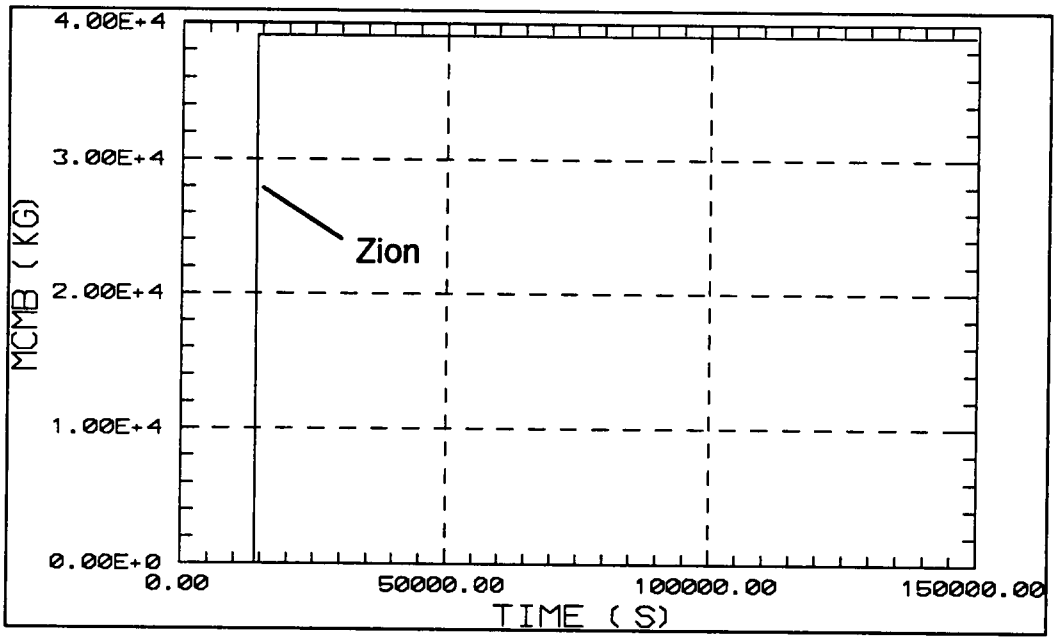

Figure 33. Debris Mass in the Lower Compartment (MCMB) for Loss of Decay Heat Removal Sequence

The water remaining in the RCS and the accumulators enters the cavity and cools the debris until it is steamed away. The debris temperature increases after the water is gone and reaches melting temperature. The debris then cools as it interacts with the concrete in the cavity and the decay power level decreases with time.

The water level in the APWR design decreases rapidly after reactor pressure vessel failure due to the large quantity of debris, relative to the Zion-like design, present in the cavity (Figure 34). Another factor is that the water injected into the cavity is less for the APWR than for the Zion-like design. This is due to the actuation of the SDV system.

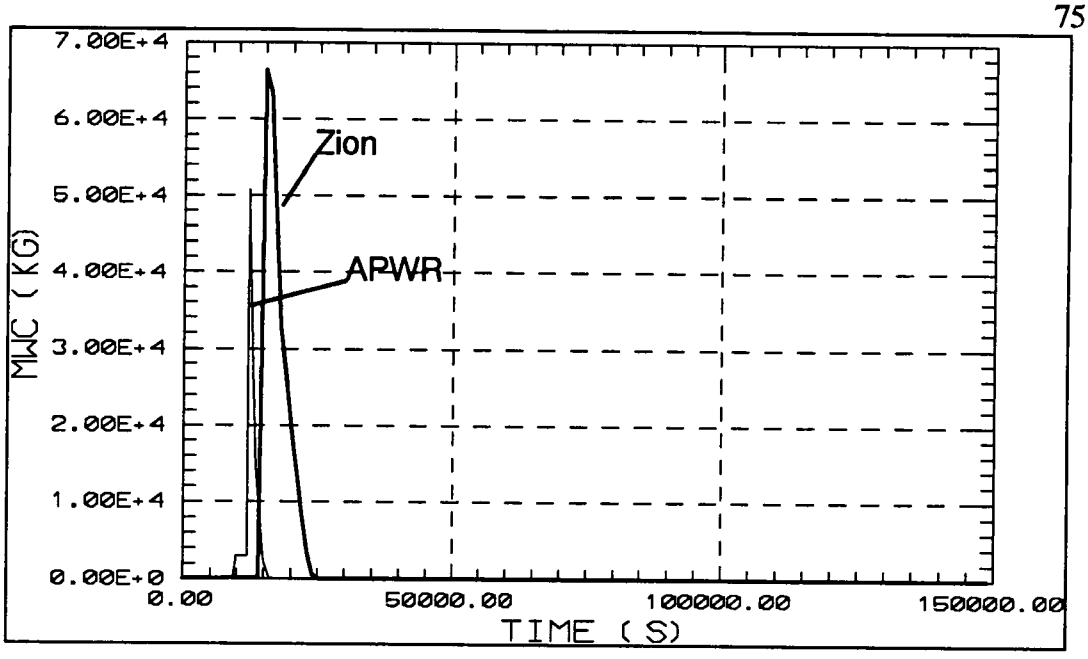

Figure 34. Water Mass Present in the Cavity (MWC) for Loss of Decay Heat Removal Sequence

The SDV system allows some of the RCS inventory, trapped in the RCS as steam for the Zion-like design, to be lost to the in-containment refueling water tank. It also allows the accumulators to dump prior to reactor pressure vessel failure and the water to be boiled away prior to reactor pressure vessel failure. These two factors result in a smaller quantity of water being present for debris cooling in the cavity.

In both cases the water inventory is boiled away and concrete attack begins. The ablation rate (Figure 35) for the APWR case is slower than for the Zion-like design due a larger cavity floor area which is required by the EPRI Requirements Document. The increased volume allows the debris to spread over a greater area and reduces the average heat flux. The lower heat flux corresponds to a slower concrete ablation rate.

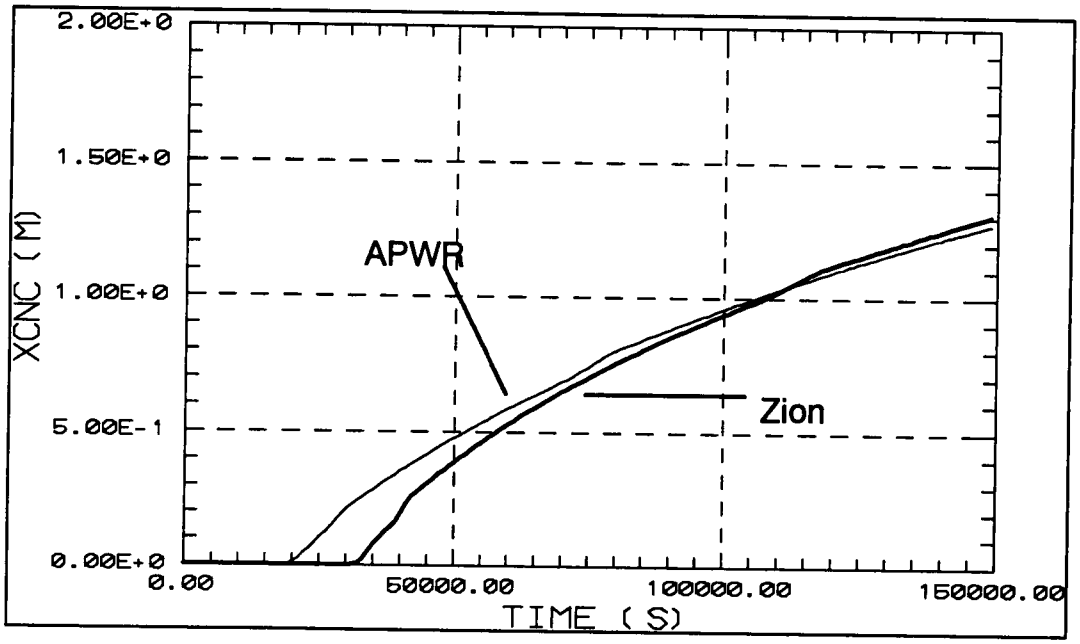

Figure 35. Concrete Ablation Depth in Cavity (XCNC) for Loss of Decay Heat Removal Sequence

The hydrogen mass in the upper containment, illustrated in Figure 36, is initially less for the APWR design than for the Zion-like design due to lower in-vessel hydrogen production. After reactor pressure vessel failure, hydrogen production is limited by ex-vessel production mechanisms, i.e., concrete attack. The presence of water in the debris cools the debris and precludes concrete attack for both cases. The quantity of hydrogen present for the APWR case decreases once the debris temperature is above 1900K due to automatic ignition of hydrogen by the hot debris. As the hydrogen contacts the hot debris the hydrogen is burned. This is possible because the steam concentration is very low and the additional production of hydrogen due to concrete attack creates locally high hydrogen concentrations suitable for burning. When the debris temperature drops, automatic ignition ceases and the hydrogen generation increases until it is near that for the Zion-like design. This is somewhat deceptive since the flattening of the curve for the Zion-like design is due to containment failure.

At the end of the analysis the hydrogen concentration in the APWR containment is 8% which is equal to the lower limit for complete hydrogen combustion to occur. If oxygen and an ignition source are present, this concentation is sufficient for global hydrogen burning to occur in dry air. The APWR containment steam concentration is in excess of 55% for this case. At this steam concentration no burning is possible and the containment is inert.

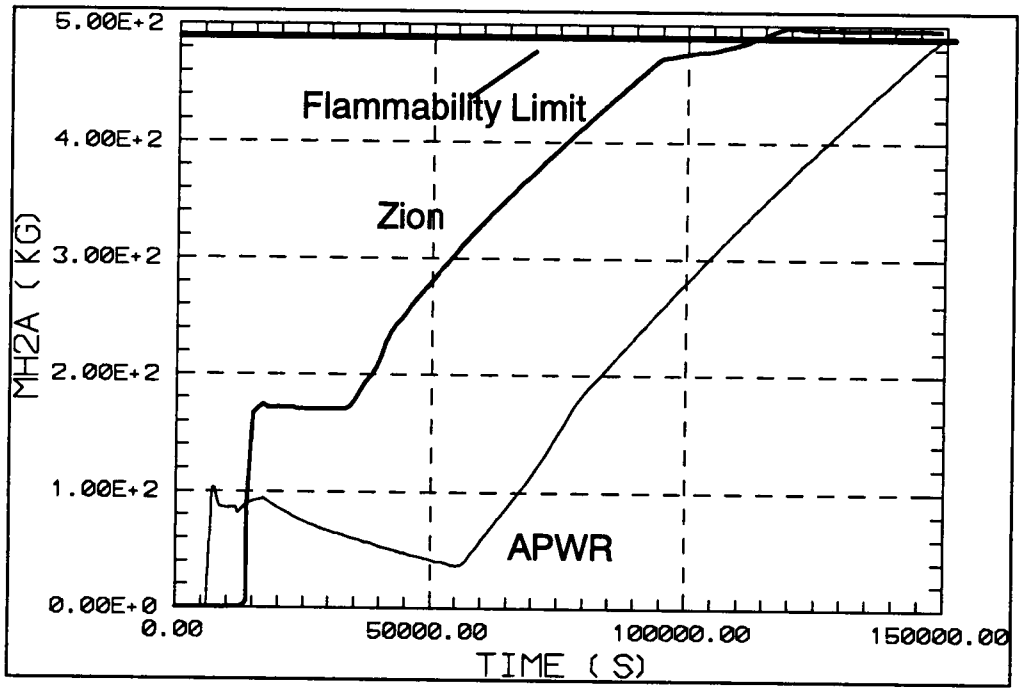

Figure 36. Hydrogen Mass in Upper Compartment (MH2A) for the Loss of Decay Heat Removal Sequence

After containment failure the containment volume is exchanged with the environment and hydrogen is lost. A significant point occurs at about 95,000 seconds. This is at the point of containment failure and the hydrogen mass present in the Zion-like design is almost twice that found for the APWR design. Although some localized partial hydrogen burning may occur in the cavity, the APWR design reduces the potential for hydrogen burning by significantly reducing the quantity of hydrogen produced.

The last parameter examined is the containment pressure. Figure 37 presents the pressure trace for the two containment designs. The containment pressure for the APWR design remains low throughout the event. A small rise occurs when the reactor pressure vessel fails. The pressure increases slowly as energy and gases are released during concrete attack. The Zion-like design pressure reaches the failure limit about 95,000 seconds (30 hours) into the analysis. As the highlighted area shows, a sudden pressure increase occurs at reactor pressure vessel failure. This is a result of high pressure melt ejection. Pressure increases as water in the cavity is boiled off away. Concrete attack provides the source of gases and energy for later pressurization until the failure pressure is reached. After failure, the pressure drops as gases trapped in the containment escape to the environment.

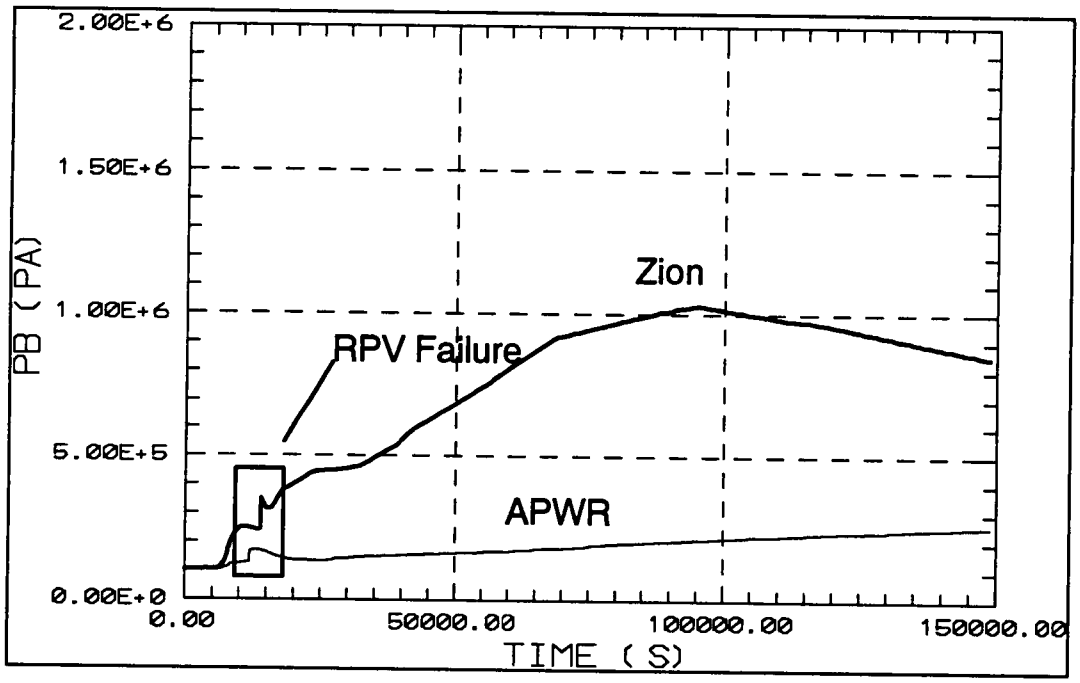

Figure 37. Lower Compartment Pressure (PB) for the Loss of Decay Heat Removal Sequence

The core damage phase of the event is similar to the station blackout case for the APWR design and the initiation of the SDV system results in the RCS pressure being lower at the time of reactor pressure vessel failure. The low pressure allows the accumulators to inject water prior to reactor pressure vessel failure and to cool the core for a short period of time. This time could be used to recover failed equipment and arrest the event prior to reactor pressure vessel failure. SDV actuation and lower RCS pressure are also advantageous in precluding the effects of high pressure melt ejection.

The removal of RCS inventory reduces the steam available for clad oxidation and reduces the total quantity of hydrogen produced in-vessel. An adverse effect of actuating the SDV system is that reactor pressure vessel failure occurs earlier in the sequence. This somewhat counteracts the effects of the accumulators and provides less time for accident mitigation and recovery. Delaying the actuation of the SDV system can offset this effect at the cost of increased hydrogen production and a potentially higher RCS pressure at reactor pressure vessel failure.

The containment analysis indicates that the operation of the cavity flooding system provides some benefit for this sequence. The timing of core damage, however, is not that different from the Zion-like design and the improvement is small. The lower RCS pressure at reactor pressure vessel failure does prevent significant quantities of debris from leaving the cavity and the sudden increase in containment pressure seen in the Zion-like design is not present for the APWR.

Although noticeable, relatively small increase in pressure for the Zion-like design would be much larger if the quantity of debris assumed to be fragmented is increased. MAAP assumes a baseline of 3% fragmentation. Some analyses have suggested that up to 50% is possible in the extreme.[23] Thus, improvements in arresting the debris within the cavity could have a much greater impact if different assumptions are made. The analysis, however, does demonstrate the potential for reducing containment pressurization during high pressure melt ejection.

Both of these benefits delay the time until containment failure and improve the chance for mitigation of the accident. The additional water provided over the debris in the cavity does increase containment pressure early in the event. This is offset, however, later in the sequence by reduced gas production from concrete attack such that the long term effect is small.

The concrete ablation depth for both cases is essentially equivalent even though the time that the debris is covered is much less and the quantity of debris in the cavity is much greater for the APWR design. The larger floor area required in the APWR design allows the debris to spread over a greater area. This spreads the heat load over a greater area and, although increasing the surface area affected, reduces ablation depth. The spreading of the debris over a large area also increases any cooling from gases flowing into the cavity and above the debris.

Finally, the decrease in hydrogen concentration and the low containment pressure provide insights into APWR hydrogen considerations. Since the SDV system discharges to the in containment refueling water tank, very little steam is present in the containment. This precludes pressurization due to steam which prolongs the time until failure.

Conversely, the lower steam concentration has a negative effect on hydrogen control. As hydrogen is produced, a lower concentration can result in a hydrogen burn. This is shown by the hydrogen burning which occurs in this sequence. If larger quantities of hydrogen are present, such that global burning is possible, the lack of steam would be a disadvantage and other hydrogen control measures, such as ignitors, could be necessary.

4.3 Small LOCA without Injection Sequence

Small LOCA sequences challenge both the decay heat removal and safety injection systems. The pipe break in the RCS results in a loss of RCS inventory. As inventory is lost the effectiveness of the decay heat removal decreases. The core uncovers after sufficient inventory is lost and core damage occurs.

The break in the RCS releases steam and energy into the containment which increases containment pressure until the containment sprays actuate to limit containment pressure. When decay heat removal is significantly reduced, the APWR design opens the SDV system valves to reduce RCS pressure. Prior to reactor pressure vessel failure, the APWR actuates the cavity flooding system

Figure 38 provides a summary of in-vessel hydrogen production. Hydrogen production is initiated approximately the same time for both designs. The larger RCS inventory present in the APWR provides some additional time prior to reactor pressure vessel failure but results in additional hydrogen generation. Figure 39 presents the collapsed water level in the reactor pressure vessel.

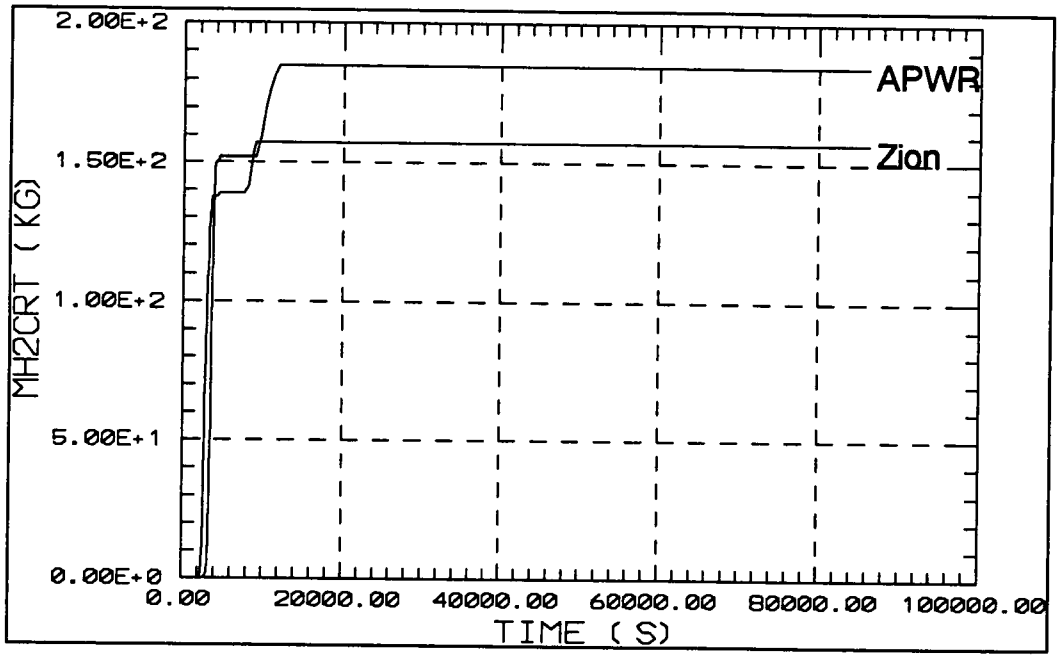

Figure 38. In-Vessel Hydrogen Generation (MH2CRT) for Small LOCA Sequence

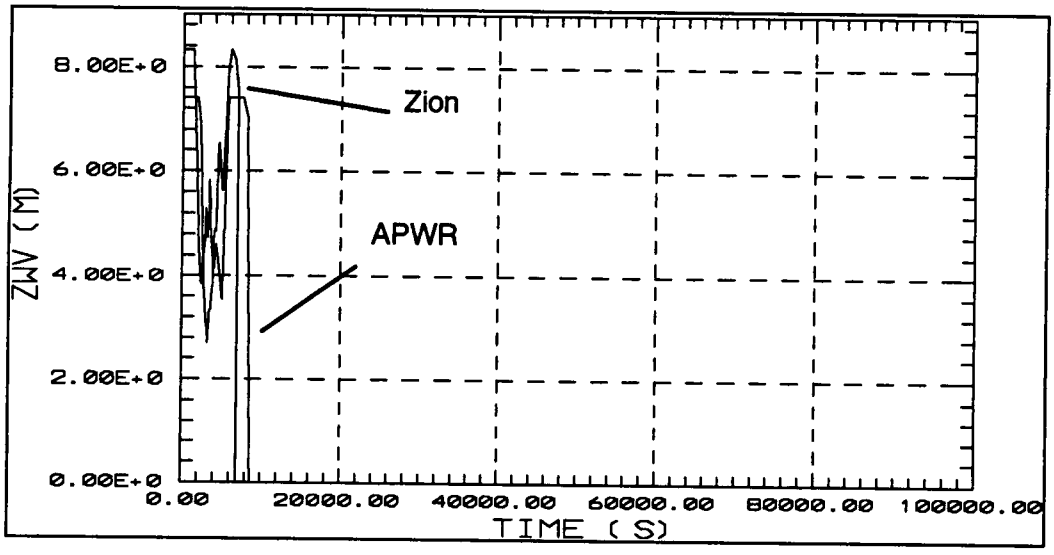

Figure 39. Collapsed RCS Water Level (ZWV) for Small LOCA Sequence

The water level for the APWR decreases more sharply due to the action of the SDV system. Since the SDV essentially changes the size of the LOCA, the peak experienced during accumulator reflood is somewhat less than that for the Zion-like plant. The larger accumulator volume present in the APWR delays reactor pressure vessel failure by maintaining the recovered level longer than for the Zion-like design.

The peak temperature, shown in Figure 40, is essentially the same for the two cases and is slightly separated by time. The peak temperature for the APWR case is somewhat lower due to the lower decay heat present later in the event. Both curves show how the temperature increases as voids occur and the brief cooling caused by the accumulators.

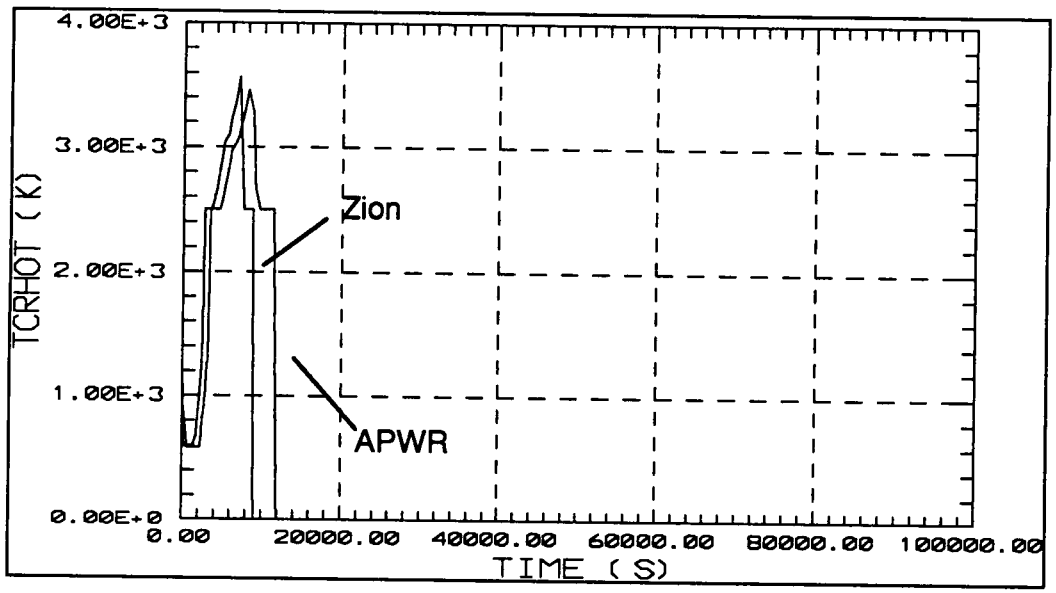

Figure 40. Peak Core Temperature (TCRHOT) for Small LOCA Sequence (Note: the temperature is recorded as zero after reactor pressure vessel failure)

The RCS pressure, Figure 41, for the two cases is also very similar. The APWR maintains a slightly higher pressure prior to the accumulators injecting. This is due to the slightly larger volume of water present in the APWR which slows the voiding process prior to the operation of the SDV system. Pressure spikes are found for both cases which are a result of the initial accumulator inventory flashing to steam when it interacts with the hot core. The opening of the SDV valves allows more accumulator inventory to initially enter the core and a higher peak pressure is reached. After the accumulators are depleted the pressures are very similar.

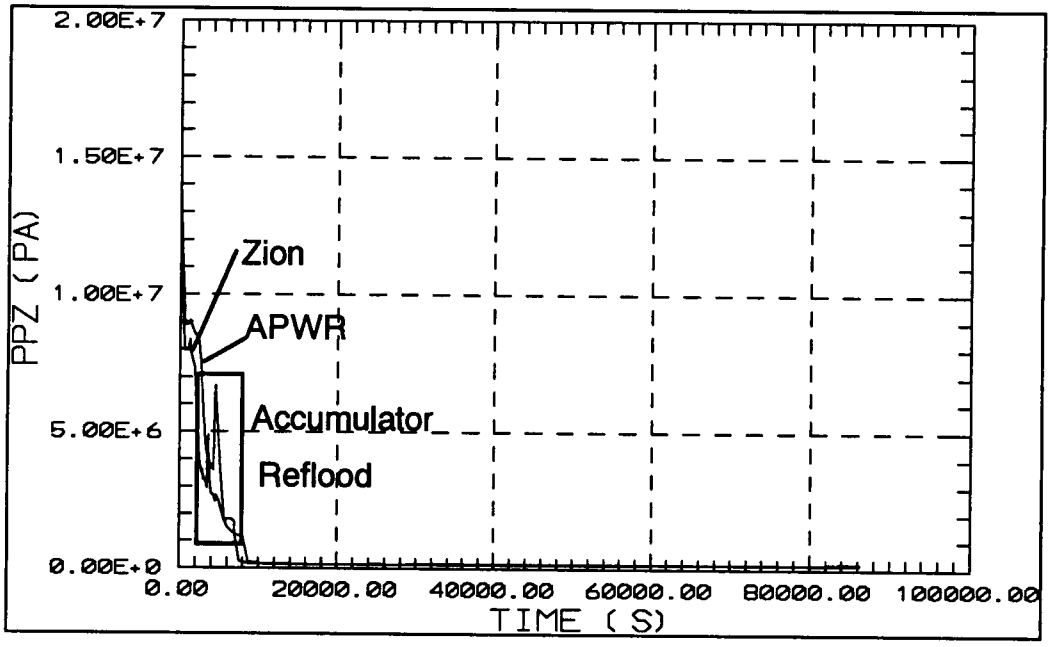

Figure 41. RCS Pressure (PPZ) for Small LOCA Sequence

The containment response for this sequence is quite different from the prior cases. The operation of the containment sprays allows for cavity flooding in the Zion-like design.

Because the APWR design is configured to promote water entering the in-containment water refueling storage tank the water, the water released by the sprays does not enter the cavity and the water present is boiled away.

This is clearly shown by examining Figure 42 and Figure 43. Figure 42 shows the average temperature of the debris in the cavity. A comparison of the early temperatures shows how the debris is quenched initially by the water present. After a series of small peaks as additional debris enters the cavity, the Zion-like design experiences low debris temperatures and the debris is essentially solidified. The debris is cool for the APWR case until the water from the cavity flooding system is boiled away. At that point the debris reheats and approaches the melt temperature. When this occurs, concrete ablation starts as shown by Figure 43.

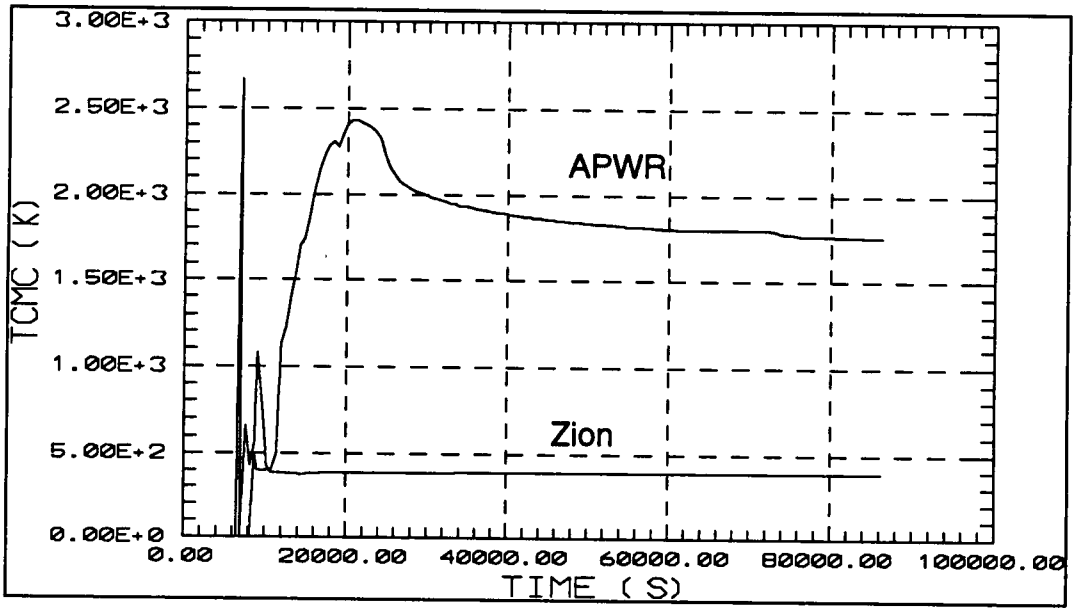

Figure 42. Average Debris Temperature in the Cavity (TCMC) for Small LOCA Sequence (Note: the temperature is recorded as zero prior to debris entering the cavity)

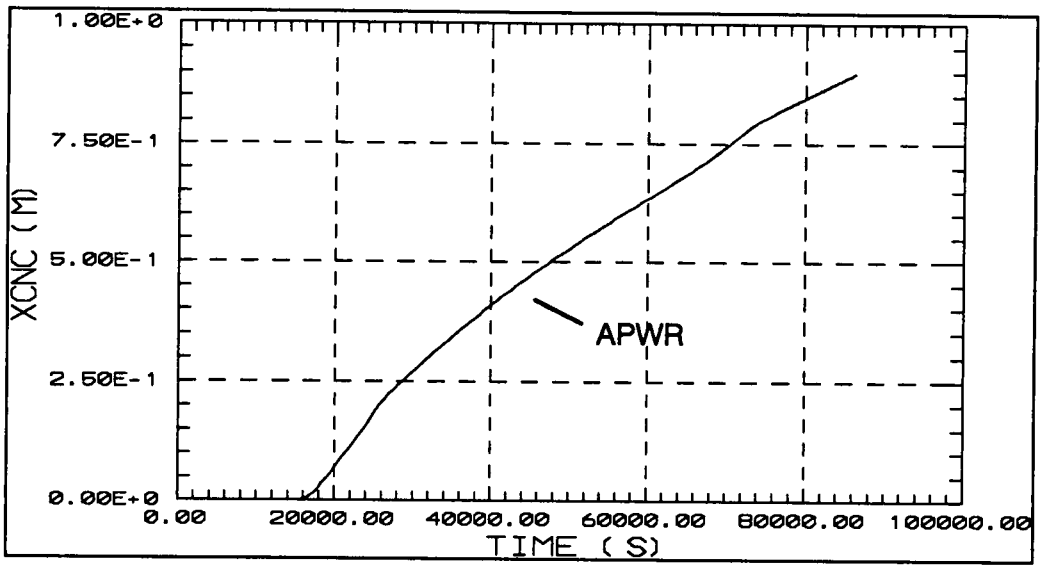

Figure 43. Concrete Ablation Depth in the Cavity (XCNC) for Small LOCA Sequence

The hydrogen present in the upper compartment is presented in Figure 44. The hydrogen concentration for the Zion-like design is dominated by in-vessel production. Very little ex-vessel generation occurs due to the rapid quenching of the debris in the cavity.

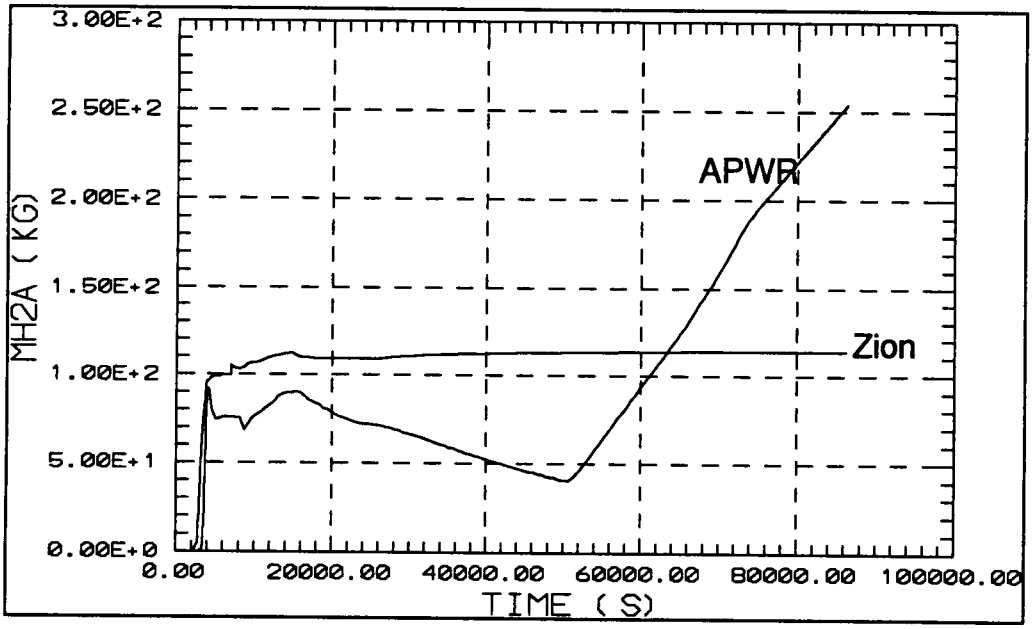

Figure 44. Upper Compartment Hydrogen Concentration (MH2A) for Small LOCA Sequence

For the APWR design, the low steam concentration and uncovered debris result in two partial burns early in the sequence and burning of hydrogen in the cavity until the debris temperature is below that required for ignition. After this point the hydrogen concentration quickly rises as a result of debris-concrete interaction and is about 2.5 times that found for the Zion-like design at the end of the analysis.

Figure 45 presents the time history of containment pressure for both cases. The Zion-like design initially has a higher pressure due to the release of steam into a smaller containment volume. The fan coolers reduce pressure slowly until the reactor pressure vessel failures. At this point a step increase is found which initiates containment sprays. The containment sprays reduce and maintain containment pressure until the RWST is empty. The pressure rises until an equilibrium is reached between the heat source and the available heat sinks.

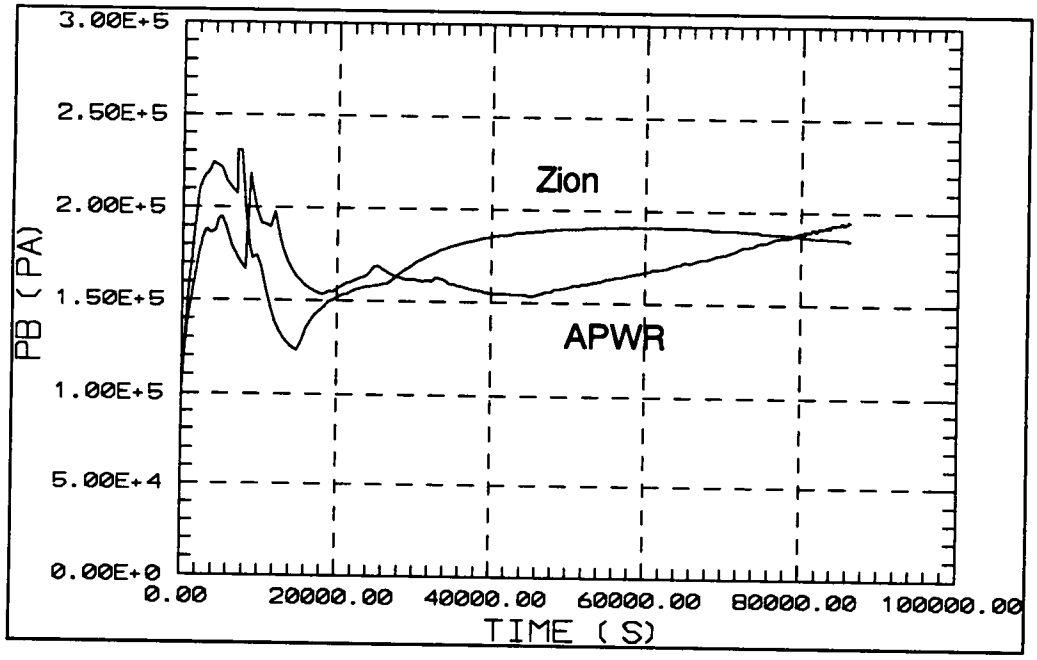

Figure 45. Lower Compartment Pressure (PB) for Small LOCA Sequence

The APWR design is similar with a lower peak pressure at reactor pressure vessel failure. The APWR containment sprays are supplied by the in-containment refueling water tank and the sprays run throughout the sequence. The pressure begins to increase about 45,000 seconds into the sequence. This is due to energy and gases generated during concrete ablation. The reaction between the debris and concrete releases noncondensable gases which fill the containment volume. Unlike steam, these gases are not condensed from the containment volume and, therefore, containment pressure cannot be reduced by the containment sprays.

The core damage phase of the event is very similar for both designs with the additional inventory present in the APWR slightly delaying core damage. RCS pressure at reactor pressure vessel failure is low for both cases due to the RCS pipe break and the actuation of the SDV system does not dramatically change the sequence.

The containment sprays and fan coolers maintain a low steam concentration for both designs and high containment pressure is not present. The operation of the containment sprays provides the Zion-like design with a means to flood and replenish the cavity and serves a similar function to the cavity flooding system found in the APWR design. The quenching of the debris highlights the importance of debris cooling and providing water to the cavity.

The concrete ablation depth for the APWR design highlights the importance of supplying water to the location of the debris to promote cooling. If the ablation continues, basemat failure may occur. The analysis assumes that a fixed inventory is available for cavity flooding. This is based on operational concerns and is further discussed in section 4.4. If additional water is provided in the APWR design the debris would remain covered and no concrete attack would occur.

Finally, the increase in hydrogen concentration and the low containment pressure provide insights into APWR hydrogen considerations. Since the containment spray system is functioning, very little steam is present in the containment. This in combination with the large amount of hydrogen generated ex-vessel providing the opportunity for a late hydrogen burn which could challenge the containment. Once again, providing additional water to the cavity and maintaining debris cooling would reduce hydrogen generation and the concern for late hydrogen burns.

Conversely, the lower steam concentration has a negative effect on hydrogen control. As hydrogen is produced, a lower concentration can result in a hydrogen burn. This is shown by the hydrogen burning which occurs in this sequence. If larger quantities of hydrogen are present, such that global burning is possible, the lack of steam would be a disadvantage and other hydrogen control measures, such as ignitors, could be necessary.

4.4 ALWR Model Sensitivity Studies

This section discusses the results of the sensitivity studies performed to address important issues related to the baseline results. The station blackout sequence was chosen for this assessment and a listing of the sensitivity studies can be found in Table 16. Comparison of different designs is used to determine the impact of the cavity flooding and SDV systems on plant performance.

The first sensitivity case compares the case with the cavity flooding and SDV systems failed to the case where both are successful. The purpose of this sensitivity study is to identify the effect of other changes in the advanced PWR containment design that improve overall severe accident response. In addition to the specific systems highlighted in this study the Requirements Document specifies other characteristics that improve performance.

Table 16
Sensitivity Studies

Case	Description
S1	Station blackout sequence with both systems functioning successfully
S2	Station blackout sequence without cavity flooding system or SDV depressurization
S3	Station blackout sequence with successful cavity flooding and no SDV depressurization

Table 17 provides a comparison of results for selected parameters. As shown, the sensitivity case resulted in additional hydrogen being generated in-vessel. The absence of the SDV system allowed larger quantities of steam to remain in the RCS to promote oxidation. The RCS pressure at the time the reactor pressure vessel fails is also higher which will increase the potential for high pressure melt ejection.

Table 17
Comparison of APWR Baseline Design to Design without SDV or Cavity Flooding Systems

Parameter	S1	S2
In-vessel hydrogen production (kg)	225	280
RCS pressure at failure (Pa)	1.4E+7	1.7E+7
Concrete ablation depth (m)	0.4	0.8
Containment pressure (Pa)	1.0E+6	2.0E+5

The unavailability of the cavity flooding system shortens the time until the debris in the cavity begins to attack the concrete. This significantly increases the amount of concrete basemat ablated by the debris following reactor pressure vessel failure. The absence of water in the cavity prevents large amounts of steam from being generated for the sensitivity case. As a result, the containment pressure is lower. As discussed earlier, pressurization due to noncondensable gases, as is the case for the sensitivity study, is much harder to control than for cases involving steam. The combination of lower steam

generation and higher hydrogen generation for the sensitivity case indicates a higher potential of hydrogen burning.

Summarizing, the presence of the two systems tends to reduce early hydrogen generation, lower RCS pressure prior to reactor pressure vessel failure, and decrease the potential for concrete attack. These all combine to reduce the potential for containment failure. The additional steam generated as a result of water covering the debris in the cavity does produce steam which tends to increase containment pressure which is an undesirable effect.

This is somewhat offset, however, by the higher steam concentrations within the containment precluding hydrogen burning. Reducing the high containment pressure due to steam is also much easier than for the case with noncondensable gases. The overall effect of the systems in relation to severe accident response is positive.

The next sensitivity case involves the cavity flooding system and examines the effect of including a cavity flooding system into the advanced PWR design. A separate dc power system with long-term capacity is assumed to be provided for this system that allows it to function following a station blackout. A comparison of concrete ablation depth for the case with and without the cavity flooding system is shown in Table 18. As shown, concrete attack is delayed by several hours by the presence of water in the cavity for the case with cavity flooding.

Table 18
Comparison of Designs with and without Cavity Flooding Systems

Parameter	S2	S3
Initiation of concrete attack (hrs)	6.94	10.28
Time to deplete water (hrs)	5.56	7.5
Containment pressure (Pa)	1.8E+5	1.0E+6

The inclusion of this system is not entirely positive. For cases without containment heat removal the successful operation of the cavity flooding system results in steam generation and containment pressurization. This is in contrast to sequences in which both the cavity flooding and containment spray systems do not function. For these sequences, the containment pressurization is due to noncondensable gas generation and the time to containment failure is much longer. This contrast represents a potentially significant decrease in the time available for recovery and the prevention of containment failure. The containment pressure for the case with successful cavity flooding is much higher at 12 hours into the event due to the additional steaming. Examing the pressurization rate later in the event indicates the rates are essentially equivalent. Once the water over the debris is gone, the sequence acts much like a case without cavity flooding and pressurization is due to noncondensable gases.

Thus, the cavity flooding system does delay core concrete attack. For sequences involving containment spray failure, however, the successful operation of the cavity flooding system may result in an increase in the likelihood of containment failure due to the reduction in the time available for recovery.

Other sensitivity studies were considered to examine important assumptions made with regard to the APWR design but were not executed due to limitations imposed on this study due to a lack of detail or restrictions within the MAAP code. These sensitivity studies are discussed below.

The amount of water available for the cavity flooding system is a major consideration in precluding debris cooling. The analysis limits the available water. If more water is available or if a continual flooding is assumed the time to boil off the water covering the

debris would be greatly increased and the hydrogen production seen in the small LOCA case would not occur.

An operational concern plays an important role in determining the amount of water provided by the cavity flooding system. If the cavity flooding system water source is not limited and it actuates during normal operation the potential exists to have water come in contact with the external surface of the reactor pressure vessel. Should this occur, a prolonged outage might be necessary to verify that the reactor pressure vessel has not be adversely affected. On the basis of this concern the EPRI Requirements Document specifies a maximum of 60,000 gallons be initially provided and that the total quantity, when transferred into the cavity, must not be capable of wetting the bottom of the reactor pressure vessel. The analysis assumes that 60,000 gallons is available.

The ability to model the IRWST using the available version of MAAP is also a concern. The present model uses the quench tank volume to represent the IRWST volume as a discharge point for the SDV system. The cavity flooding system is modeled using a special system model designed to model an ex-containment cavity flooding system. These two models are not connected and no feedback is provided. As a result, water from the IRWST cannot be lost to the cavity flooding system as would occur in the actual APWR design. This is not a major concern as long as the quantity of water used in cavity flooding is small compared to the IRWST volume.

The present assumption results in about a 10 to 1 ratio which is considered sufficiently large to prevent alterations in the expected results. If a larger volume is assumed for the cavity flooding system this would no longer be the case and the potential exists for artificially changing the analysis results. For example, the containment pressurization caused by steaming would be increased due to the added water inventory

in the containment. If a more recent version of MAAP were available, this concern could be overcome and additional sensitivity studies could be performed.

Another important assumption is the time at which the SDV system is actuated. The manual actuation of this system will be based on the status of certain plant parameters. These parameters have not been identified and will probably be vendor specific. Procedures for current PWR designs are used as a surrogate for identifying when the SDV system will be actuated. Current loss of decay heat removal procedures require that the operators depressurize the RCS when a temperature criterion, typically 1200F as measured by the core exit thermocouples, is reached. This criterion is used in the analysis.

Actuating the SDV system at this time results in an early loss of RCS inventory which corresponds to core damage occurring earlier for some sequences in the APWR design than the Zion design. If the SDV system is actuated later in the sequence the time to core damage will be delayed. Other parameters, however, may not be favorably affected. Hydrogen production would increase proportionally to account for the longer residence time of water in the RCS. The RCS pressure at reactor pressure vessel failure may also be higher if actuation of the SDV system is postponed. Varying the actuation time has been previously examined by another analysis[22] and is not repeated here.

Finally, the analysis assumes that 3% of the debris initially expelled at reactor pressure vessel failure is finely fragmented and available for DCH. Larger amounts have been considered possible in other studies[30] with a reasonable upper bound being 50%. Addressing the effect of this uncertainty on the results was attempted. Fragmentation values of 5% and 10% were attempted using the station blackout sequence with no significant change in containment pressurization. Analysis of other current PWR designs

for the IPE have arrived at similar conclusions and indicate that noticeable changes do not occur at fragmentation values of less that 25%.[31,32] Attempts at higher fragmentation values were not possible due to internal restrictions in the version of the MAAP code used for the analysis. More recent versions of the MAAP code have removed this restriction and could further investigate how increasing this parameter would change the overall analysis.

CHAPTER 5

CONCLUSIONS AND RECOMMENDATIONS

The MAAP code has been successfully ported to an IBM 386 personal computer. An on screen interface has been created and added to the MAAP code to provide the user with a convenient means of observing how the execution of the MAAP code is progressing. The personal computer version of the MAAP code has been successfully benchmarked by comparing results with the mainframe version provided certain time step constraints are met.

Information contained in the EPRI Requirements Document, supplemented by other sources, has been used to establish an advanced PWR plant design which reflects the intent of the Requirements Document. This design, as represented by the MAAP parameter file, has been evaluated over a range of postulated accident sequences to examine the improvement in severe accident response afforded the APWR design features. In particular, the SDV and cavity flooding features are considered. The results of the analysis have been compared with those for a Zion-like plant design which is representative of a present day design.

The results indicate that the APWR design is somewhat more robust with regard to severe accident challenges. Specifically, the design configuration results in lower in-vessel hydrogen generation and a higher degree of ex-vessel debris cooling. The use of the cavity flooding system to ensure that water is present in the cavity does result in higher containment pressure due to steam generation for cases without containment heat

removal. The higher pressures can, over time, result in containment failure occurring earlier than for similar cases assessed for the Zion-like design.

The large amounts of steam, however, do tend to inert the containment which precludes hydrogen burning and are much easier to control than the pressurization mechanism present in the Zion-like design. Overall, changes in the APWR design result in a better overall response. Although the analysis results presented here have examined a wide range of conditions and provide a good basis for evaluating the APWR plant response, additional work should be performed to refine the present analysis and are discussed below.

The MAAP code is constantly evolving with several revisions occurring during the development of this report. The most recent version, MAAP 4, provides significant improvement in several key areas assessed in this report. This includes better core relocation models and a more complex model for assessing high pressure melt ejection. Assessing how the fraction of debris fragmented following reactor pressure vessel failure effects the overall analysis is an example of where the most recent version could improve the analysis. The execution of similar sequences using this newer version could result in a more accurate picture of plant performance.

This report was performed using the most recent versions of the EPRI Requirements Document which were available at the time of the analysis. The development of the EPRI Requirements Document is an ongoing process and additions to or modifications of the present requirements could alter the results of this analysis. Once the requirements document is completed, a final analysis should be performed to finalize the conclusions of this study.

Due to the lack of design detail, several assumptions were made that introduce uncertainties. Therefore, the results should not be used to determine whether any specific APWR design will meet the severe accident mitigation requirement. Additional analysis and sensitivity studies using plant-specific design information could be performed if the information was available. This is especially true for assumptions on the amount of water available for cavity flooding system and the timing of SDV actuation. The present analysis identifies both positive and negative aspects of the design using the present assumptions. Design information for a specific design would provide clearer insight into these features and would lead to a clearer conclusion as to their effectiveness. This would strengthen any overall statement that the APWR design is an improvement over present designs in relation to severe accident mitigation.

Although these improvements could be used to further improve this analysis, sufficient modeling and analysis has been performed to gain insights into the advanced design. The results and the insights obtained demonstrate the robustness of the Requirements Document in providing guidance for addressing severe accident challenges and in improving plant performance.

LIST OF REFERENCES

LIST OF REFERENCES

1. Electric Power Research Institute (EPRI), Advanced Light Water Reactor Utility Requirements Document, Chapter 1: Overall Requirements, Volume II, Revision 4, April 1992.

2. U. S. Nuclear Regulatory Commission (USNRC), Policy Statement on Safety Goals for the Operation of Nuclear Power Plants, Federal Register, Vol. 51, p. 28044, August 4, 1986.

3. Technology for Energy Corporation (TEC), Industry Degraded Core Rulemaking (IDCOR) Program Plan, November 1981.

4. TEC, Industry Degraded Core Rulemaking (IDCOR) Technical Summary Report, November 1984.

5. USNRC, Individual Plant Examination for Severe Accident Vulnerabilities, 10CFR50.54(f), Generic Letter 88-20, November 23, 1988.

6. Fauske & Associates, Inc. (FAI), MAAP 3.0B – Modular Accident Analysis Program for LWR Power Plants, Volume 1: User Guidance, and Volume 2: Code Structure and Theory, EPRI NP–7071–CCML, November 1990.

7. Fontana, M. H. A. R. Buhl, and R. D. Copp, "The DOE Advanced Reactor Severe Accident Program (ARSAP) - An Overview," Proceedings of the Conference on Probabilistic Safety Assessment and Risk Management, PSA 87, Zurich, Switzerland, August 30 - September 4, 1987, Verlag TUV Rheinland GmBH, Koln, 1987.

8. Boyd, G.J. S.R. Lewis, R.L. Summitt, R.D. Brooks, Interim Probabilistic Assessment of the EPRI Requirements for Advanced Pressurized Water Reactors, ARSAP Technical Report DOE/ID-10207, Safety and Reliability Optimization Services (SAROS), Incorporated and International Technology (IT) Corporation, July 1988.

9. Summitt, R. L., Probabilistic Assessment of the EPRI Requirements for Evolutionary Pressurized Water Reactors Supplement 1 - Containment Analysis, DOE/ID-10268, Advanced Reactor Severe Accident Program (ARSAP), Safety and Reliability Optimization Services (SAROS), Inc., August 1990.

10. ABB/Combustion Engineering, System 80+ PRA, June 1992.

11. Summitt, R.L. M.H. Fontana, "Primary Coolant Depressurization for Minimizing Direct Containment Heating", Proceedings of the OECD/NEA CSNI Specialist Meeting on Intentional Coolant System Depressurization, Garching, Federal Republic of Germany, June 12-14, 1989, Gesellschaft fur Reaktorsicherheit mbH, Garching, 1989.

12. FAI, Technical Support for the EPRI Debris Coolability Requirement for Advanced Light Water Reactors, ARSAP, June 1990.

13. FAI, Technical Support for the Hydrogen Control Requirement for the EPRI Advanced Light Water Reactor Requirements Document, ARSAP, January 1990.

14. Cathcart, J. V., et al., "Zirconium Metal-Water Oxidation Kinetics IV. Reaction Rate Studies," ORNL/NUREG-17, August 1977.

15. Baker, L. Jr., et al., "Studies of Metal-Water Reactions at High Temperatures III. Experimental and Theoretical Studies of the Zirconium-Water Reaction," ANL-6548, May 1962.

16. Frid, W., Containment Severe Accident Thermohydraulic Phenomena, RAMA III, Final Report, Swedish Nuclear Power Inspectorate, RAMA III 89-04, August 1991.

17. Microway, Inc., NDP FORTRAN-386, User's Manual, Version 1.1, 1988.

18. Electric Power Research Institute, Advanced Light Water Reactor Utility Requirements Document, Chapter 5: Engineered Safety Systems, Volume II, Revision 4, April 1992.

19. Electric Power Research Institute, Advanced Light Water Reactor Utility Requirements Document, Chapter 6: Building Design and Arrangement, Volume II, Revision 4, April 1992.

20. Allen, M. D., et. al., Experiments to Investigate the Effect of Flight Path on Direct Containment Heating (DCH) in the Surtset Test Facility, NUREG/CR-5728, October 1991.

21. USNRC, Reactor Risk Reference Document, Appendices J-O, Draft for Comment, NUREG-1150, January 1987.

22. FAI, Technical Support for Issue Resolution, Technical Report 85.2, IDCOR Program, July 1985.

23. Carter, J.C., III, R.L. Summitt, R.E. Henry, et al., Prevention of Early Containment Failure due to High Pressure Melt Ejection and Direct Containment Heating for Advanced Light Water Reactors, ARSAP, July 1990.

24. USNRC, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, NUREG-1150, June 1989.

25. Summitt, R.L. EPRI Requirements Document DCH Depressurization Valve Sizing Study, ARSAP Task 8.3.5, January 1988.

26. Pickard, Lowe, and Garrick, Inc., Zion Probabilistic Safety Study, September 1981.

27. FAI, Approximate Source Term Methodology for Pressurized Water Reactors, FAI-85-58, March 1987.

28. Electric Power Research Institute, Analysis of Three Mile Island - Unit 2 Accident, NSAC-1, March 1980.

29. Malinovic, B. and Sharon A., Simulation of Phases 3 and 4 of the TMI-2 Accident Using MAAP-DOE and MAAP-3.0B, DOE/ID-10259, ARSAP, Fauske and Associates, Inc., April 1990.

30. Westinghouse Electric Corporation, Background Information for Westinghouse Owner's Group Emergency Response Guideline, FR-H.1, Response to Loss of Secondary Heat Sink, LP-Rev. 1, September 1983.

31. Summitt, R. L., SHNPP PRA: Section 8.0, Containment Response Assessment, SAROS, Inc., December 1993.

32. Summitt, R. L., H. B. Robinson PRA: Section 8.0, Containment Response Assessment, SAROS, Inc., September 1993.

SUPPLEMENTAL BIBLIOGRAPHY

American Nuclear Society and the National Institute of Electrical and Electronic Engineers, PRA Procedures Guide - A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants, Final Report, NUREG/CR-2300, January 1983.

Epstein, M. and Fauske, H. K., "The Three Mile Island Unit 2 Core Relocation-Heat Transfer and Mechanism", Nuclear Technology 87, December 1989.

Electric Power Research Institute, Hydrogen Combustion in Reactor Containment Buildings, Technical Report 12.3, Industry Degraded Core Rulemaking (IDCOR) Program, September 1983.

Harper, F. T., et al., Evaluation of Severe Accident Risks: Quantification of Major Input Parameters, Volume 2, Rev. 1, Part 2, U. S. Nuclear Regulatory Commission (USNRC), NUREG/CR-4551, Sandia National Laboratories, December 1990.

Mendoza, Z. T., et al, Generic Framework for IPE Back-End (Level 2) Analysis, Volume 1: Main Report, NSAC-159, Nuclear Safety Analysis Center, October 1991.

Mendoza, Z. T., et al, Generic Framework for IPE Back-End (Level 2) Analysis, Volume 2: PWR Implementation Guidelines, NSAC-159, Nuclear Safety Analysis Center, October 1991.

Tarbell, W. W., et al, Pressurized Melt Ejection into Water Pools, NUREG/CR-3916, Sandia National Laboratories, March 1991.

Yang, J.W., PWR Dry Containment Issue Characterization, NUREG/CR-5567, Brookhaven National Laboratories, August 1990.

Appendix

MAAP Screen Program Listing

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C    SUBROUTINE SCREEN
C
C            MAAP SCREEN PROGRAM REVISION 2.1
C            WRITTEN BY: RICKY LYNN SUMMITT
C
C        COPYRIGHT 1990, ALL RIGHTS RESERVED
C
C    THE CODE AUTHOR REQUESTS TO BE NOTIFIED OF ANY
C    MODIFICATIONS TO THIS CODE
C
C    ************************************
C    FOR COMPATIBILITY WITH THE MOST RECENT VERSION OF MAAP
C    INSERT THE MOST RECENT COMMON BLOCK STATEMENTS FROM
C    THE MAIN HERE.
C
C    FAILURE TO USE THE MOST RECENT BLOCK STATEMENTS CAN
C    RESULT IN VARIABLE RELOCATION AND UNPREDICTABLE RESULTS
C    ************************************
     IMPLICIT REAL  (A-H,K-Z)
     SAVE
     REAL*8 TIMD,TIMFPD
C    THIS TITLE BLOCK MUST BE ADDED AFTER MAIN COMMON BLOCKS
C    ARE ADDED TO THIS SUBROUTINE IN ORDER TO HAVE TITLE
C    IT IS FOUND IN OUTPT1 SUBROUTINE
C
     COMMON/TITLE/ITITL(80,3),IBLANK
C

COMMON/CONSTS/MOLWW,MOLWH2,MOLWO2,MOLWCO,MOLWC2,MOLWN2,MOLWCA,
    1CVH2,CVO2,CVCO,CVC2,CVN2,CPH2,CPO2,CPCO,CPC2,CPN2,
    2GAMMA,RGAS,GRAV,PI,SIGMA,TCRITW,PCRITW,LHICE,CPICE,
    3MOLWU2,MOLWZR,MOLWZO,MOLWCS,TU2MP,TZRMP,TZOMP,TCSMP,
    4QMH2O2,QMCAC2,QMZRW,QMCAW,QMZRC2,QMCOO2,MWMIN,QMCSW,QMCSCO,
    5FCNCTL,DUM73,MOLWSO,MOLWCI,TCAMP,IDIAG
     COMMON/MODEL/FRCOEF,ACFSTR,TTRX,ACFPR,FAOX,TCFLAM,FFPREL,HTFB,
    1HTSTAG,FAOUT,FWETBW,HTCMCR,FCMDA,FCMDCH,CDA,CDB,CDC,CDD,CDU,
    2FCHTUR,FDROP,FSPAR,FVOL,TTENTR,EW,EWL,EEQ,ECM,EG,FCRHY,FENTR,
    3FFLOOD,FCHF,NVP,FCDBRK,SCALU,SCALH,GSHAPE,CSHAPE,FAERDC,FDF1,
    4FEFFDR,FCSIVP,FEMBRT,FCDDC,TCLMAX,LHEU,VFCRCO,XRSEED,FPRAT,
    5FTEREL,TEU,FFRICR,FNCBP,VFCRBL,NSAMP,FFRICX,FXEGAP,VFSEP,
    6TJBRN,FASI,FASR,FABA,FAKO,FCRDR,FROUPZ,FCRBLK,FE0,FNUDRP,FMOD
     COMMON/CNCRT/CPCN0,TCNMP,LHDEC,LHCN,MFCN(12),DCSRCN,
    2PETEN,PEREB,PEPTEN,PEPREB,PSSPH,PSSPZ,
    2PSSYHT,PSSYHR,PSSFHT,PSSFHR,PEL,PEPL,PSSYHL,PSSFHL
     COMMON/CNCR2/LHRCN
     COMMON/CONTRL/IPSPR,ICPR,IRUNG,IRSTW,IRSTR,IHUW,ICDUM,IPOUT,
    1ICOUT,IPLT1,ISUM,IPTSAV,IPTSPK,IPTSMX,IRECIR,IDISCH,IBW,
    1IAUXW,IAUXR,INODRB,IFPRAT,ITEREL,ISTW,ISTR,IBATCH,IPAINT,
    1INPGRP,IMAX1,IMAX2,IRMAX,JCHMAX,IHUR,INBB,IMXFP1,IMAX3,IMAX4,
    2INNODE,INUB,IINX(14),IIN2X(14),IBRKX(14),INSR,ICWRI,IDON(22),
    3IOUT2X(14),INSMGR,INPHSF(15),JBH,INLOOP,JDC,JUH,JDM,JBB,JUB,JVP,
    4JZCBL,JZSG,JUP,IDONRB(30),ICWALL(2,12),IMXFP2,ILCM,ILEST,
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5IUO2,IZR,IZO,ICS,ISO,INSB,INRBC,INFPMX,ITL
 COMMON/EVENTC/IEVNT(250),IAEVMX,IEVMAX,JEVNT(240),JCOUNT,ISUDDN,
1JSTART,IOLD(250)
 COMMON/FP1/MFPIN(25),MSM0(5),FQP(12)
 COMMON/FP2/QGFPA,QGFPB,QGFPC,QGFPD,QGFPI,
1QGFPU,QGFPQT,QGFPPZ,QGFPPS(14),QGFPBS,QWFPA,QWFPB,QWFPC,QWFPD,
2QWFPI,
1QWFPU,QWFPQT,QWFPPZ,QVCR(70),QCMFPP,QCMFPC,QCMFPA,
2QCMFPB,LAMSA,LAMSB,LAMSC,LAMSD,LAMSI,LAMSU,MAWEA,MAWEB,MAWEC,
3MAWED,MAWEI,MAWEU,QFPHSF(19),QFPBHF(3),MFP0(12)
 COMMON/PS1/NCL,XDHL,XRRV,VLOWCR,ACR,
1VCL1,XRVP0,QCP0,WWMU0,TWMU,XDCL,ZSR,FBB,ABB,ZBB,MSGPHD,
2VBLKCL,VCL,VHL,VRV,WGRXV0,VDC,VLOWDC,FSR,NHL,VFCPMX,PPSL,
3PPSH,TDIFPS,ZWPZL,ZWPZH,TDSCRM,ZWSGL,NWPST,WPST(5),TIWPST(5),
4ZSGTS,XTRV,ZNOZ,ZLOWNZ,VHL1,WWLET0,PDCR0,FUB,AUB,ZUB,
5ZDMP,ZDM,ZFLANG,ADMWX,MCBL,MCBH,MUPI,MDMP,MDMW,MHLTOT,MCLTOT,
7MRV,AGUP,XDHUP,AUPI,QC0,FINPS,XTINS,ZRVCYL,VHD,ACBL,ACBH
 COMMON/PS2/PPS,TCMPS,TWPS,PPSTPS,TGPS,
1QCMTPS,HWPS,VLOWNZ,TCRHOT,HCMPS,VEXPS,MFSTUP,WWDCCR,
2MGPS,WH2PLV,WH2PVA,TIFRRX,VWPS,ZWBH,VFPS,MFSTPS,MFH2PS,
3VPS,TWUI,WSTCMP,VWBI,VWUI,TWBI,WWUL,CFPS,QWDCCR,GWBKMX,TWHPS,
4TWLPS,HTCPPS,VOLGPS,TICPUL,WSTCRI,WH2CRI,QGCRI,FTGPS,QHYDRO,
5MFSTDR,MWPS,UWPS,ZWBC,ARV,VUPDM,ZWCPS,MFCSPS,TWSTPS,
6VGPS,MH2PS1,MDWTOT,MCMTPS,TICPBL,MH2CB1,HWSTPS,HSTSTP,VSTPS,
7TCMMPP,TIRXFL,MFZRPS,MFU2PS,MFZOPS,ZWUC,DCMPS,ZSEDBB,VSCMPS,
8QGHSPS,KCMPS,ADC,USTPS,MFO2PS,MFCOPS,MFC2PS,MFN2PS,MUGPS,CPGPS,
9KGPS,DIFPS,VFUL,WWBL,ZWUH,ZWDC,MWBH,MWUH,ACRMIN,CVWPS,HSTDR,VFCR,
1TWUT,VWUT,TWBT,VWBT,WHLBL,WHLUL,WSGBL,WSGUL,TSGBHP,HWBT
 COMMON/HEAT1/XDPIN,MZR0,NPIN,MU20,ZCRL,ZCRU,TIRRAD,QCR0,NCHAN,
2NAXNOD,FPA(20),FPR(7),FA(7),EXPO,FALPHA,ENRCH,FCR,FQFISS(3),
3FXOX0,XRPEL,ABP,XTZR,VOLGPN,PFILL,MCSP0,FVLGP,FVUGP,XRBF,AGBF,
4AGCRFP,NCRFP
 COMMON/HEAT2/QDECAY,WSTVOI,HWCR,WSTCR,WH2CR,QGCR,WU2CR,WZRCR,MCR,
2WWUPCR,VWCR,QWUP,MH2CR1,MWCR1,WSTCNT,HWUP,WFLDC,QWCR,TWCR,TWDC,
3ZWV,PDCR,TISCRM,TDHU,TCRN(70),TGCHMX(9),WFQCCR,
4WZOCR,QCMCR,VOLCR,WSTCH(9),WH2CH(9),ZSUBCR,DWSTCR,WGUPCR,
5QCRBF(5)
 COMMON/PZ1/VPZ,APZ,PPZHT0,PPZSP0,ZPZHT,QPZHT0,WPZSP0,
1WPZSV0,PPZSVL,PPZSVH,XDSR,ZPZSP,XLSR,NPZSV,XDPZSP,PPORVL,
2PPORVH,NPORV,WPORV0,MPZ,FLSEAL,ASEDPZ,PDEADS,PDEADP,MPZSR
 COMMON/PZ2/HWPZ,HSTPZ,VGPZ,TGPZ,TWPZ,PPZ,ZWPZ,VWPZ,APZSV,PPSTPZ,
1WPZSP,VEXPZ,VFPZ,TPZSAV,SHLPZ,FTGPZ,TSR1,MFH2PZ,MFSTPZ,VOLGPZ,
2VEXEQ,APORV,APZX,MH2PZ1,MFO2PZ,MFCOPZ,MFC2PZ,MFN2PZ,VWDC,WWPZMU,
3MUGPZ
 COMMON/ES1/XDACUM,PLPI0,PHPI0,PACUM0,TRWST,TACUM,
1MRWST0,MACUM0,ARWST,XLACUM,PSP0,PFAN0,NFN,WVFN0,XDSP,
2VACUM,NACUM,NHPI,NLPI,NHPIPT,ZHDHPI(5),WVHPI(5),
3NLPIPT,ZHDLPI(5),WVLPI(5),PCHP0,NCHP,NCHPPT,ZHDCHP(5),WVCHP(5),
4ACS,ZCS,NSPPT,ZHDSP(5),WVSP(5),ZHDRCH(5),ZHDRLP(5),ZHDRHP(5),
5ZHDRSP(5),NSPA,NSPB,ZESPRW,ZESPCS,ZESPSI,WSPNZ0,PSPNZ0,WRWSTX,
6TDHPI,TDLPI,TDCHP,TDSPA,TDSPB,TDFAN,NTFC,ATFC,AFINFC,FFINFC,
7RGFLHX,XDFNFC,XTTFC,KTFC,AFLMNF,XIDTFC,NREGFC,TCWHX,WCWFC,NLPSP0,
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     8FFCDB,FHXSP,NTSP,NBSP,XIDTSP,XTTSP,XTCSP,XSSP,KTSP,XBCSP,XSTSP,
     9WCWSP,ZWPZMU,FHXRH,NTRH,NBRH,XIDTRH,XTTRH,XTCRH,XSRH,KTRH,XBCRH,
     1XSTRH,WCWRH,NTUSP,NTURH,XIDSSP,XIDSRH,MUHI0,XLUHI,XDUHI,PUHI0,
     2VUHI,PDUHI,MWCIT0,WWCI0,WLPIX,WHPIX,WCHPX,WSPAX,WLPSPX,WSPBX,
     3NAFWPT,WVAFW(5),ZHDAFW(5),ACST,ZCSTAF,ZSGAFW
      COMMON/ES2/WESFHL,HACUM,PACUM,WESFDC,TSPA,TSPB,WSPTA,WSPTB,WCS,
     1WESFCL,VWACUM,PSRCHP,PSRLPI,PSRHPI,PSRSPA,PSRSPB,ZWRWST,
     2TIHPI,TILPI,TICHP,TISPA,TISPB,TIFAN,WCDFC,QCDFC,WSTFN,WH2FN,
     3WO2FN,WCOFN,WC2FN,WN2FN,QGFN,PSSI,PSSP,PSLPI,NLPISI,VWSI,
     4VWLPI,PSATSI,WWUHI,PUHI,PSATLP,PSATSP,WLPSPT,TLPSP,QESFCL,QESFHL,
     5QESFDC,WWCI,HWCI,PSHUTO,PSRLPS
      COMMON/SG1/VSG,ASGDC,AFLWSG,ZAFW,MWCST0,ZWSG0,TFW,PSGSVL,
     1PSGSVH,NSGSV,WSGSV0,PSGRV,NSGRV,WSGRV0,WFWMX,VSGPHD,TDAFW,
     2TDMSIV,VSGPRI,XTSGTS,TAFW,NTSG,XTSG,XIDSG,KTSG,WAFWXU,FARVBX,
     3FARVUX,ZWCTLB,ZWCTLU,XDSGTS,ZDEADB,ZDEADU,WBATMN,FMSLB,XLSHEL,
     4MSHEL,FINSG,NSB,ASB,ZSB,ASEDSG,WAFWXB,AMSLB
      COMMON/SG2/ZWBS,QWBSC,PBS,TGBS,QSGTOT,WCDHBS,QCDHBS,ZWUS,QWUSC,
     1PUS,TGUS,HWBS,WCDHUS,QCDHUS,WWFWBS,WWFWUS,TIAFW,TIMSIV,ZTAVSG,
     2WCDCBS,QCDCBS,WCDCUS,QCDCUS,QWBSH,QWUSH,
     3TWBS,TWUS,WSTBSB,WSTUSB,HSTBS,HSTUS,VGUPCR,VEXBS,
     4VOLGBS,VGBS,SHLBSF(3),MUGBS,VWBS,WGBST,WSGSIE,ASGSV,ASGRV
      COMMON/CAV1/ATNBP,APLC,ACHARC,ZV,ATN0,AKEY,VC0,ZTN,
     1AOWC,XLOWC,RGOWC,XOWC0,KOWC,CPOWC,DOWC,NIGC,XIGC,ASEDC,ATNEX
      COMMON/CAV2/PC,TCMC,TWC,ZWC,TGC,PPSTC,HWC,HSTC,TIXBC,MAFLC,
     1WH2PLC,HCMC,MWC1,HWSTC,TIFRC,VGC,DCMC,ZWBUC,ATN,VWC,VFC,
     2TFMC,WFLVP,WNMOLC,CFC,VOLGC,MFSTC,MFH2C,MFO2C,MFCOC,MFC2C,MFN2C,
     3MH2C1,MCMTC,XCNC1,MGC,STCMC,FTGC,
     4TCMMPC,WSTSC,USTC,HTCPC,VSTC,DUM5,WCDOWC,SHLOWC,MUGC,
     6WFPRC(12)
      COMMON/QT1/VQT,MWQT0,PQTRD,ZQTRD,ASEDQT
      COMMON/QT2/PQT,WSTRD,WH2RD,QGRD,PPSTQT,WWRD,QWRD,TWQT,TGQT,
     1VWQT,MN2QT,VOLGQT,MH2QT1
      COMMON/B1/ZSPB,ACMPLB,ZCURBB,ACHARB,VB0,ZDBC,ZDAB,XTGAP,
     2AOWB,XLOWB,RGOWB,XOWB,KOWB,CPOWB,DOWB,FOWCSB,AIWB,XLIWB,
     3RGIWB,XIWB0,KIW,CPIW,DIW,AFB,XLFB,RGFB,XFB,KFB,CPFB,DFB,MEQB,
     4AEQB,HTOW,DUM66,FABIR,FAADR,ABD,ABA,NIGB,XIGB,ZFB,ASEDB,ZEQB
      COMMON/B2/TGB,PB,HSTB,VGB,TIXBB,WH2PLB,TIFRB,QCMTB,TWB,TCMB,
     1PPSTB,HWB,ZWB,XPTN,AWLB,MFSTB,VWB,MFH2B,MFO2B,MFCOB,MFC2B,
     2MFN2B,VFB,TFMB,WCDDB,QCDDB,QGDCKB,HWSTB,CFB,VOLGB,MH2B1,MCMTB,

    3XCNB1,TWSATB,MGB,QGOWB,WCDOWB,QCDOWB,FTGB,TCMMPB,HTCPB,USTB,HCMB,
     4ZCMB,VCMB,MUGB,CPGB,KGB,DIFB,VSTB,CPSTB,WCDPB,SHLOWB,RAB0,
     5WVSWB,WSTB,WFPRB(12)
      COMMON/E2/PE,VOLGE,MFSTE,MFH2E,MFO2E,MFCOE,MFC2E,
     2MFN2E,VGE,TGE,HSTE,MAWE,MSWE,PBKDIS,HWSTE
      COMMON/A1/VA,ARP,ZSPA,AAD,ACHARA,ZCURBA,AOWA,XLOWA,RGOWA,XOWA,
     1KOWA,CPOWA,DOWA,FOWCSA,AIWA,XLIWA,RGIWA,XIWA,ADCK,XLDCK,RGDCK,
     2XDCK,KDCK,CPDCK,DDCK,FDCKCS,MEQA,AEQA,NIGA,XIGA,NIGAB,NIGAD,ZA,
     3FSPA,FWRPB,PCF,FCFA,XRCONT,ALKNOM,MWBAG0,ASEDA,
     4NHOOPA,XTREHA,XTREZA,XDHOPA,ZACYL,ZIWA,XDZFA,XDRFA,ZEQA
      COMMON/A2/TGA,PA,TIXBA,VGA,HSTA,HWA,ZWA,MFSTA,MFH2A,MFO2A,
     1MFCOA,MFC2A,MFN2A,TWA,PPSTA,TFMA,CFA,VOLGA,MH2A1,MGA,WGFNA,FTGA,
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2TCMMPA,TCMA,XCNA1,MCMTA,USTA,HTCPA,TSDCKA,VSTA,WCDPA,SHLOWA,
3MUGA,HWSTA,WVSWA,XDISRA,XDISZA,STRVA,WFPRA(12)
 COMMON/D1/VD,ADPL,ZDDB,ACHARD,AOWD,XLOWD,RGOWD,XOWD,KOWD,CPOWD,
1DOWD,FOWCSD,ZCURBD,NIGD,XIGD,ASEDD,AIMP,XDIMP,AGRATE,
2NHOOPD,NTENZ,XTREHD,XDHOPD,XDTENZ,XTREZD,XDZFD,XDRFD
 COMMON/D2/TGD,TIXBD,HSTD,VGD,PD,ZWD,PPSTD,TFMD,CFD,VOLGD,MFSTD,
1MFH2D,MFO2D,MFCOD,MFC2D,MFN2D,TWD,MH2D1,MGD,FTGD,USTD,HTCPD,VSTD,
2WCDPD,SHLOWD,MUGD,HWSTD,XDISRD,XDISZD,STRVD,ACF
 COMMON/I1/VI,TEXITI,TWICE,VWICE,ASUMPI,ZSUMPI,ZI,ABI,ASEDI
 COMMON/I2/PRI,TGI,ZWI,VWI,TWI,CFI,
1VOLGI,MFSTI,MFH2I,MFO2I,MFCOI,MFC2I,MFN2I,VGI,HSTI,VSTI,MICE1,
2MH2I1,MGI,FTGI,USTI,HTCPI,WCDICE,PPSTI,MUGI,HWSTI
 COMMON/U1/VU,ACHARU,ZU,AUA,NIGU,XIGU,ZUA,ASEDU
 COMMON/U2/TIXBU,TFMU,TWSATU,HSTU,VGU,TGU,PU,HWU,MFSTU,MFH2U,MFO2U,
1MFCOU,MFC2U,MFN2U,CFU,VOLGU,PPSTU,MH2U1,MGU,FTGU,USTU,HTCPU,MUGU,
2HWSTU
 COMMON/CUPL1/HWAC,HSTVP,HWVP,TGVP,MFH2VP,MFSTVP,VGVP,HSTCB,TGCB,
1MFSTCB,MFH2CB,MFO2CB,MFCOCB,MFC2CB,MFN2CB,HWTN,VWA,QGSRPS,QWSRPS,
2TGSR,MFH2SR,MFSTSR,VGSR,VWSR,VOIFSR,HSTBB,TGBB,HWBB,QSTB,
4MFSTBB,MFH2BB,HCMTN,MFO2BB,MFCOBB,MFC2BB,MFN2BB,
6MFO2VP,MFCOVP,MFC2VP,MFN2VP,MFO2SR,MFCOSR,MFC2SR,MFN2SR,TWBB,
7HSTUB,TGUB,MFSTUB,MFH2UB,MFO2UB,MFCOUB,MFC2UB,MFN2UB,
8HWUB,TWUB,MFSTCF,MFH2CF,MFO2CF,MFCOCF,MFC2CF,MFN2CF,VGCF,
9TGCF,HSTCF,HWSTCF,MFSTLK,MFH2LK,MFO2LK,MFCOLK,MFC2LK,MFN2LK,VGLK,
9TGLK,HSTLK,HWSTLK
 COMMON/CUPL2/MFSTBD,MFH2BD,MFO2BD,MFCOBD,MFC2BD,MFN2BD,HSTBD,
1TGBD,HWI,MFSTBI,MFH2BI,MFO2BI,MFCOBI,
2MFC2BI,MFN2BI,TGBI,HSTBI,TGIU,HSTIU,MFSTIU,MFH2IU,
3MFO2IU,MFCOIU,MFC2IU,MFN2IU,TSPAB,TGBA,HSTBA,
4MFSTBA,MFH2BA,MFO2BA,MFCOBA,MFC2BA,MFN2BA,HSTUA,
5TGUA,MFSTUA,MFH2UA,MFO2UA,MFCOUA,MFC2UA,MFN2UA,
6HWDB,MFSTAD,MFH2AD,MFO2AD,MFCOAD,MFC2AD,MFN2AD,HSTAD,TGAD,
1MFSTTN,MFH2TN,MFO2TN,MFCOTN,MFC2TN,MFN2TN,HSTTN,TGTN,
3HWSTCB,HWSTBA,HWSTBD,HWSTAD,HWSTTN,HWSTBI,HWSTIU,HWSTUA,
4MFSTSB,MFH2SB,MFO2SB,MFCOSB,MFC2SB,MFN2SB,HSTSB,TGSB,HWSB
 COMMON/THFLOW/WGRV,WWRV,WWSR,WGSR,WWPZDR,WGPZDR,WWSB,
1WGSB,WWVP,WGVP,WWBB,WGBB,WWUB,WGUB,WGRXV,WWRXV,
1WGCFA,WGCFD,WGLK,WWTN,WGCB,WGBD,WWIB,WGBI,WGIU,WWAB,WGBA,
1WWAC,WGUA,WWUI,WWDB,WGAD,WWSPAB,WGTN,
1WCMAC,WCMVP,WCMTN,
1WAWCF,WSWCF,WAWLK,WSWLK,
3WSWCB,WSWBA,WSWBD,WSWAD,WSWTN,WSWBI,WSWIU,WSWUA,
4WAWCB,WAWBA,WAWBD,WAWAD,WAWTN,WAWBI,WAWIU,WAWUA
 COMMON/ICMNX1/MSTPS,MH2PS,UGPS,MU2PS,UCMPS,MZRPS,XRVP,
1MWBI,UWBI,MWUI,UWUI,MAWA,MAWB,MAWC,MAWD,MAWI,MAWU,
2MSTPZ,MH2PZ,UGPZ,MWPZ,UWPZ,TSR,
3UWB,MWB,UCMB,DUM30,DUM1,MSTB,MH2B,MO2B,MCOB,MC2B,MN2B,UGB,DUM22,
4TOWB(15),TFB(15),TIWB(15),XMIWB,XCNB,TEQB,MWC,DUM31,DUM2,UWC,
5UCMC,MSTC,MH2C,MO2C,MCOC,MC2C,MN2C,UGC,MH2BS,TOWC(15),XMOWC,XCNC,
6MSTA,MH2A,MO2A,MCOA,MC2A,MN2A,UGA,MWA,UWA,TOWA(15),TDCK(15),
7TIWA(15),TEQA,MSTD,MH2D,MO2D,MCOD,MC2D,MN2D,UGD,MWD,UWD,TOWD(15),
8MICE,MWI,MSTI,MH2I,MO2I,MCOI,MC2I,MN2I,UWI,UGI,
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9MRWST,MACUM,MWBS,MSTBS,UWBS,UGBS,MWUS,MSTUS,UWUS,UGUS,MWQT,MSTQT,
 1MH2QT,UWQT,UGQT,MWU,MSTU,MH2U,MO2U,MCOU,MC2U,MN2U,UWU,UGU,MWG,
 1MWCR,MWDC,UWCR,UWDC,UCRN(70),MZRN(70),MU2N(70),MH2CRT,
 2FQCMPS,FQCMC,FQCMB,MZON(70),MZOPS,MCOBS,MC2BS,MO2BS,MN2BS,
 5MWUP,MCSPS,MUHI,DUM50,XCRDC,XCRUC,DUM51,XCRDB,XCRUB,VCNFC,
 6MWCST,MH2CBT,XCNA,XCRDA,XCRUA,DUM52,UCMA,FQCMA,DUM9,DUM3,DUM24,
 7DUM14,DUM33,MWBT,MWUT,UWBT,UWUT
  COMMON/ICMNX2/MO2PS,MCOPS,MC2PS,MN2PS,MO2PZ,MCOPZ,MC2PZ,MN2PZ,
 1MWCIT,DUM40,DUM41,DUM42,FQNVCR(70),FECL(70),MSWA,MSWB,MSWC,
 2MSWD,MSWI,MSWU,MCSP(7),TCNA(15),TCNB(15),TCNC(15),
 3MCMC(50),MCMB(50),MCMA(50)
  COMMON/ICMNF1/FMSTPS,FMH2PS,FUGPS,FMU2PS,FUCMPS,FMZRPS,FXRVP,
 1FMWBI,FUWBI,FMWUI,FUWUI,FMAWA,FMAWB,FMAWC,FMAWD,FMAWI,FMAWU,
 2FMSTPZ,FMH2PZ,FUGPZ,FMWPZ,FUWPZ,FTSR,FUWB,FMWB,FUCMB,
 3DUM34,DUM4,FMSTB,FMH2B,FMO2B,FMCOB,FMC2B,FMN2B,FUGB,DUM15,
 4FTOWB(15),FTFB(15),FTIWB(15),FXMIWB,FXCNB,FTEQB,FMWC,DUM35,DUM61,
 5FUWC,FUCMC,FMSTC,FMH2C,FMO2C,FMCOC,FMC2C,FMN2C,FUGC,FMH2BS,
 6FTOWC(15),FXMOWC,FXCNC,FMSTA,FMH2A,FMO2A,FMCOA,FMC2A,
 7FMN2A,FUGA,FMWA,FUWA,FTOWA(15),FTDCK(15),FTIWA(15),FTEQA,
 8FMSTD,FMH2D,FMO2D,FMCOD,FMC2D,FMN2D,FUGD,FMWD,FUWD,FTOWD(15),
 9FMICE,FMWI,FMSTI,FMH2I,FMO2I,FMCOI,FMC2I,FMN2I,FUWI,FUGI,
 1FMRWST,FMACUM,FMWBS,FMSTBS,FUWBS,FUGBS,FMWUS,FMSTUS,FUWUS,FUGUS,
 1FMWQT,FMSTQT,FMH2QT,FUWQT,FUGQT,FMWU,FMSTU,FMH2U,FMO2U,FMCOU,
 2FMC2U,FMN2U,FUWU,FUGU,FMWG,FMWCR,FMWDC,FUWCR,FUWDC,FUCRN(70),
 3FMZRN(70),FMU2N(70),FMH2CR,FFQCPS,FFQCMC,FFQCMB,FMZON(70),
 4FMZOPS,FMCOBS,FMC2BS,FMO2BS,FMN2BS,
 7FMWUP,FMCSPS,FMUHI,DUM53,FXCRDC,FXCRUC,DUM54,FXCRDB,
 8FXCRUB,FVCNFC,FMWCST,FMH2CB,FXCNA,FXCRDA,FXCRUA,DUM55,FUCMA,
 9FFQCMA,DUM62,DUM72,DUM26,DUM17,DUM36,FMWBT,FMWUT,FUWBT,FUWUT
  COMMON/ICMNF2/FMO2PS,FMCOPS,FMC2PS,FMN2PS,FMO2PZ,FMCOPZ,FMC2PZ,
 1FMN2PZ,FMWCIT,DUM43,DUM44,DUM45,FFQNVC(70),FFECL(70),
 2FMSWA,FMSWB,FMSWC,FMSWD,FMSWI,FMSWU,FMCSP(7),FTCNA(15),
 3FTCNB(15),FTCNC(15),FMCMC(50),FMCMB(50),FMCMA(50)
  COMMON/TIMING/TTPRNT,TILAST,TDMAX,TDMIN,MFCHMX,MDFPMN,FTGCHX,
 1MFCHFP,TTSTAT,TIOP,TIM,TD,NCRTEQ,TDOLD,TDAUX
  COMMON/INITL/TWPS0,PPS0,ZWPZ0,PB0,TGB0,TGI0,FRHB0,MICE0,MWSG0,
 1TCN0,PSG0,FRHA0,TGA0,TWPSI,PPSI,TDIFBW
  COMMON/PS3/V(16),WSTX(16),WH2X(16),QGX(16),K(18),FV(16),Z(16),
 1AG(16),XDH(16)
  COMMON/CIRC2/VG(16),QGPSB,W(22),CPG(16),MG(16),TSATPS,HWPSP,
 2HSTPSP,DHWDP,TDC,WVUL,WVBL,FLOSS,HTPSB0,
 3HSTCR,HSTUP,HSTBH,HSTBHT,HSTBCT,HSTBIL,HSTBC,HSTDC,
 4HSTUH,HSTUHT,HSTUCT,HSTUIL,HSTUC,HSTDM,HSTC1,HSTB1,
 5APHSF(19),ASHSF(15),SHLPSF(19),ZPHSF(19),ASEDPF(19),
 6TPHSF(19),TSHSF(15),QPHSF(19),QSHSF(15),QPSC
  COMMON/ICMNXC/TGCR,TGUP,TGBH,TGBHT,TGBCT,TGBIL,TGBC,TGDC,
 1TGUH,TGUHT,TGUCT,TGUIL,TGUC,TGDM,TGC1,TGB1,
 2MFH2CR,MFH2UP,MFH2BH,MFHBHT,MFHBCT,MFHBIL,MFH2BC,MFH2DC,
 3MFH2UH,MFHUHT,MFHUCT,MFHUIL,MFH2UC,MFH2DM,MFH2C1,MFH2B1,
 4XOXN(9,16),TCBL(5,5),TPZ(5,5),TDUM(10,5),TUPI(5,2),TCBH(5,5),
 5TDMP(5,2),TBH(2,3),TBHT(5,3),TBCT(5,1),TBIL(5,3),TBC(2,3),
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     6TRV(5,5),TUH(2,3),TUHT(5,3),TUCT(5,1),TUIL(5,3),TUC(2,3),
     7TDMW(5,5),TBSS(5,5),TUSS(5,5)
      COMMON/ICMNFC/FTG(16),FMFH2(16),FXOXN(9,16),FTCBL(5,5),
     4FTPZ(5,5),FTDUM(10,5),FTUPI(5,2),FTCBH(5,5),FTDMP(5,2),
     5FTBH(2,3),FTBHT(5,3),FTBCT(5,1),FTBIL(5,3),FTBC(2,3),FTRV(5,5),
     6FTUH(2,3),FTUHT(5,3),FTUCT(5,1),FTUIL(5,3),FTUC(2,3),FTDMW(5,5),
     5FTBSS(5,5),FTUSS(5,5)
      COMMON/FOMS2/WH2BRL,WH2BRC
      COMMON /RB1/VOLRB(10),AWATRB(10),AHSRB(10),XHSRB(10),KHSRB(10),
     1CPHSRB(10),ZHSRB(10),DHSRB(10),WVORB(10),WVIRB(10),
     2ASEDRB(10),AIMPRB(10),XDIMRB(10),AGRARB(10),WSPRB(10),XHSPRB(10),
     3FIDIRB(10),FIDRRB(10),ZFRB(10),WC2RB(10),AIWRB(10),
     4XTIWRB(10),KIWRB(10),CPIWRB(10),ZIWRB(10),DIWRB(10),NHSRB(10),
     5MSPRB0,MC2RB0,TRB0,TWSPRB,XDSPRB,FRHRB,TAMB,PAMB,TFDAMP,TSPRB,
     6TC2RB,MAERFF,FDFFIL
      COMMON /RB2/QHSIRB(10),WCONIR(10),ZWRB(10),
     1VOLGRB(10),PPSTRB(10),TOHSRB(10),TFLMRB(10),HSTRB(10),
     1HTCPRB(10),USTRB(10),MUGRB(10),CPGRB(10),KGRB(10),DIFRB(10),
     1MFSTRB(10),MFO2RB(10),MFH2RB(10),MFN2RB(10),MFCORB(10),MFC2RB(10),
     1CFRB(10),XWFIRB(10),MFLRB(10),UFLRB(10),
     1QWFPRB(10),WSWRB(10),HWSTRB(10),MGRB(10),VWRB(10),MSWGRB(10),
     1LAMSRB(10),WVSWRB(10),MAWERB(10),TFDIFR(10),HWRB(10),
     1WGINRB(10),WGORB(10),WVINRB(10),FTGRB(10),QGFPRB(10),
     1VGRB(10),VSTRB(10),SHLRB(10),NFSTRB(10),NFH2RB(10),NFO2RB(10),
     1TFLRB(10),TGRB(10),PRB(10),TWRB(10),MOLWRB(10),DUMA(10),
     1HGRB(10),WRB(50),WCCRB(50),TIBRRB(10),TIXBRB(10),PDBKRB
      COMMON/THFPX/MCMPS(50)
      COMMON/THFPF/FMCMPS(50)
      COMMON/TPOLGY/IVOL(50,2),ZJUNC(50,2),IJRB(10,10),IJMAX(10),
     1AJUNC0(50),XLJUNC(50),IHRB(50),XWJUNC(50),XHJUNC(50),PFFJ(50),
     2PFBJ(50),IFRB(50),IMAXJ,IJUMAX,ZJRBC,IJACT(30),
     3IDIRB(10),IDAMP(10),IRBSP(10),IRBC2(10),IBRRB(10),INHSRB(10)
      COMMON/UNITS/IMODEL( 70),ICNCRT( 31),ICNCR2(  1),IFP1  ( 42),
     & IFP2  (151),IPS1  ( 76),IPS2  (101),IHEAT1( 64),IHEAT2(133),
     & IPZ1  ( 25),IPZ2  ( 31),IES1  (171),IES2  ( 52),ISG1  ( 44),
     & ISG2  ( 44),ICAV1 ( 19),ICAV2 ( 59),IQT1  (  5),IQT2  ( 13),
     & IB1   ( 43),IB2   ( 68),IE2   ( 15),IA1   ( 50),IA2   ( 50),
     & ID1   ( 27),ID2   ( 31),II1   (  9),II2   ( 26),IU1   (  8),
     & IU2   ( 24),ITIMIN( 15),IINITL( 16),IPS3  (146),ICIRC2(256),
     & IFOMS2(  2),IRB1  (283),
     & IUNITS
C*THCOM
      COMMON/TIME2/TIMD,TIMFPD
      INTEGER YPOS,XPOS
      CHARACTER*1 ULBR,URBR,VERTBAR,HORZBAR,CLS,LOCTXY
      CHARACTER*1 LLBR,LRBR,ESC,CUBR,CDBR,CLBR,CRBR
      CLBR=CHAR(185)
      CRBR=CHAR(204)
      CUBR=CHAR(202)
      CDBR=CHAR(203)
      ULBR=CHAR(201)
      URBR=CHAR(187)
```

```
      VERTBAR=CHAR(186)
      HORZBAR=CHAR(205)
      LLBR=CHAR(200)
      LRBR=CHAR(188)
      ESC=CHAR(27)
      TIME=0.0
      XVALUE=327.6
      WRITE(*,*)ESC//"[2J"
C
C     THIS ROUTINE DRAWS A MAAP STATUS SCREEN WHICH IS UPDATED
C     AT EACH PRINTOUT POINT
C
C     PRINT OUT TITLE BLOCK
C
      WRITE(*,*)ESC//"[1;30H",ESC//"[37;40m",
CIBM  WRITE(*,*)ESC//"[1;29H",ESC//"[37;40m",
     1ULBR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//URBR
      WRITE(*,*)ESC//"[2;30H",
     1VERTBAR
      WRITE(*,*)ESC//"[2;52H",
     1VERTBAR
      WRITE(*,*)ESC//"[3;30H",
     1VERTBAR
      WRITE(*,*)ESC//"[3;52H",
     1VERTBAR
      WRITE(*,*)ESC//"[4;30H",
     1VERTBAR
      WRITE(*,*)ESC//"[4;52H",
     1VERTBAR
      WRITE(*,*)ESC//"[5;30H",
     1LLBR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//LRBR
      WRITE(*,*)ESC//"[2;32H",
     1"MAAP RUNTIME SUMMARY"
      WRITE(*,5)ESC//"[3;32H",
     1TITLE
5     FORMAT(A10,A18)
      WRITE(*,6)ESC//"[4;32H",
     1TIMD
6     FORMAT(A10," RUNTIME= ",1PE9.3)
C
C     PRINTOUT BLOCK FOR THE CORE NODE
C
C     PRINTS THE FOLLOWING TO THE SCREEN BLOCK
C
C     HOTTEST CORE NODE TEMPERATURE
C
```

```
      WRITE(*,*)ESC//"[1;5H",ESC//"[37;40m",
     CIBM  WRITE(*,*)ESC//"[1;4H",ESC//"[37;40m",
        1ULBR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
        1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
        1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//URBR
         WRITE(*,*)ESC//"[2;5H",
        1VERTBAR
         WRITE(*,*)ESC//"[2;21H",
        1VERTBAR
         WRITE(*,*)ESC//"[3;5H",
        1VERTBAR
         WRITE(*,*)ESC//"[3;21H",
        1VERTBAR
         WRITE(*,*)ESC//"[4;5H",
        1VERTBAR
         WRITE(*,*)ESC//"[4;21H",
        1VERTBAR
         WRITE(*,*)ESC//"[5;5H",
        1VERTBAR
         WRITE(*,*)ESC//"[5;21H",
        1VERTBAR
         WRITE(*,*)ESC//"[6;5H",
        1LLBR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
        1HORZBAR//HORZBAR//CDBR//HORZBAR//HORZBAR//
        1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//LRBR
         WRITE(*,*)ESC//"[2;9H",ESC//"[31;40m",
        1"  CORE"
         WRITE(*,*)ESC//"[3;10H",
        1"HOTTEST"
         WRITE(*,*)ESC//"[4;9H",
        1"NODE TEMP"
C     PRINT OUT VARIABLES
11    FORMAT(A10,1PE9.3)
      WRITE(*,11)ESC//"[5;10H",(((TCRHOT-273.0)*(9.0/5.0))+32.0)
C
C     PRINTOUT REACTOR VESSEL BLOCK
C
C     THIS ROUTINE PRINTS OUT
C
C     REACTOR VESSEL TEMPERATURE (WATER)
C     REACTOR VESSEL TEMPERATURE (GAS)
C     MASS OF WATER IN THE VESSEL
C     REACTOR VESSEL WATER LEVEL
C
      WRITE(*,*)ESC//"[8;3H",ESC//"[32;40m",
     CIBM  WRITE(*,*)ESC//"[8;2H",ESC//"[32;40m",
        1ULBR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
        1HORZBAR//HORZBAR//
        1HORZBAR//HORZBAR//CUBR//HORZBAR//HORZBAR//
        1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//URBR
         WRITE(*,*)ESC//"[7;13H",VERTBAR
         WRITE(*,*)ESC//"[15;13H",VERTBAR
```

```
      WRITE(*,*)ESC//"[9;3H",
     1VERTBAR
      WRITE(*,*)ESC//"[9;21H",
     1VERTBAR
      WRITE(*,*)ESC//"[10;3H",
     1VERTBAR
      WRITE(*,*)ESC//"[10;21H",
     1VERTBAR
      WRITE(*,*)ESC//"[11;3H",
     1VERTBAR
      WRITE(*,*)ESC//"[11;21H",
     1VERTBAR
      WRITE(*,*)ESC//"[12;3H",
     1VERTBAR
      WRITE(*,*)ESC//"[12;21H",
     1VERTBAR
      WRITE(*,*)ESC//"[13;3H",
     1VERTBAR
      WRITE(*,*)ESC//"[13;21H",
     1VERTBAR
      WRITE(*,*)ESC//"[14;3H",
     1LLBR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//CDBR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//LRBR
      WRITE(*,*)ESC//"[9;7H",ESC//"[33;40m",
     1"RX VESSEL"
      WRITE(*,*)ESC//"[10;5H",
     1"WTEMP."
      WRITE(*,*)ESC//"[11;5H",
     1"GTEMP."
      WRITE(*,*)ESC//"[12;5H",
     1"MWTR"
      WRITE(*,*)ESC//"[13;5H",
     1"RX WTL"
      WRITE(*,11)ESC//"[10;12H",(((TWCR-273.0)*(9.0/5.0))+32.0)
      WRITE(*,11)ESC//"[11;12H",(((TGCR-273.0)*(9.0/5.0))+32.0)
      X=(MWCR-MWBH-MWUH)
      WRITE(*,11)ESC//"[12;12H",(X*2.2)
      WRITE(*,11)ESC//"[13;12H",(ZWV/0.3048)
C     WRITE(*,*)ESC//"[8;32H",ESC//"[37;40m",
      WRITE(*,*)ESC//"[14;24H",ESC//"[33;40m",
CIBM  WRITE(*,*)ESC//"[14;23H",ESC//"[33;40m",
     1ULBR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//CUBR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//CUBR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
```

```
     1HORZBAR//HORZBAR//URBR
      WRITE(*,*)ESC//"[16;13H",CRBR//
     1HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//CLBR
      WRITE(*,*)ESC//"[15;24H",
     1VERTBAR
      WRITE(*,*)ESC//"[15;72H",
     1VERTBAR
C     WRITE(*,*)ESC//"[16;24H",
C    1CLBR
      WRITE(*,*)ESC//"[16;72H",
     1VERTBAR
      WRITE(*,*)ESC//"[17;24H",
     1VERTBAR
      WRITE(*,*)ESC//"[17;72H",
     1VERTBAR
      WRITE(*,*)ESC//"[18;24H",
     1LLBR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//CDBR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//LRBR
      WRITE(*,*)ESC//"[15;36H",ESC//"[32;40m",
     1"PRIMARY SYSTEM"
      WRITE(*,*)ESC//"[16;28H",
     1"PRESS."
      WRITE(*,*)ESC//"[16;48H",
     1"TEMP."
      WRITE(*,*)ESC//"[17;28H",
     1"H2 MASS"
      WRITE(*,*)ESC//"[17;48H",
     1"     "
      WRITE(*,11)ESC//"[16;37H",(PPS/6804.0)
      WRITE(*,11)ESC//"[16;57H",(((TWPS-273.0)*(9.0/5.0))+32.0)
      WRITE(*,11)ESC//"[17;37H",(MH2CRT*2.2)
CSP   WRITE(*,11)ESC//"[17;57H",MFCSIP
C
C    PRINTOUT PRESSURIZER DATA BLOCK
C
C     THIS ROUTINE PRINTS OUT
C
C     PRESSURIZER TEMPERATURE (WATER)
C     PRESSURIZER PRESSURE
C     PRESSURIZER WATER LEVEL
C     HYDROGEN GENERATION IN THE PRESSURIZER
```

```
C
      WRITE(*,*)ESC//"[7;28H",ESC//"[35;40m",
     CIBM WRITE(*,*)ESC//"[7;27H",ESC//"[35;40m",
     1ULBR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//URBR
C    WRITE(*,*)ESC//"[14;13H",VERTBAR
      WRITE(*,*)ESC//"[8;28H",
     1VERTBAR
      WRITE(*,*)ESC//"[8;46H",
     1VERTBAR
      WRITE(*,*)ESC//"[9;28H",
     1VERTBAR
      WRITE(*,*)ESC//"[9;46H",
     1VERTBAR
      WRITE(*,*)ESC//"[10;28H",
     1VERTBAR
      WRITE(*,*)ESC//"[10;46H",
     1VERTBAR
      WRITE(*,*)ESC//"[11;28H",
     1VERTBAR
      WRITE(*,*)ESC//"[11;46H",
     1VERTBAR
      WRITE(*,*)ESC//"[12;28H",
     1VERTBAR
      WRITE(*,*)ESC//"[12;46H",
     1VERTBAR
      WRITE(*,*)ESC//"[8;30H",
     1" PRESSURIZER"
      WRITE(*,*)ESC//"[9;30H",
     1"TEMP."
      WRITE(*,*)ESC//"[10;30H",
     1"PRES."
      WRITE(*,*)ESC//"[11;30H",
     1"WTRL."
      WRITE(*,*)ESC//"[12;30H",
     1"H2MASS"
      WRITE(*,*)ESC//"[13;28H",ESC//"[35;40m",
     CIBM WRITE(*,*)ESC//"[13;27H",ESC//"[35;40m",
     1LLBR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//CDBR//HORZBAR//
     1HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//LRBR
      WRITE(*,11)ESC//"[9;37H",(((TWPZ-273.0)*(9.0/5.0))+32.0)
      WRITE(*,11)ESC//"[10;37H",(PPZ/6804.0)
      WRITE(*,11)ESC//"[11;37H",(ZWPZ/0.3048)
      WRITE(*,11)ESC//"[12;37H",(MH2PZ*2.2)
C
C    PRINTOUT SAFETY INJECTION SYSTEM BLOCK
C
C    THIS ROUTINE PRINTS OUT
```

```
C
C   SAFETY INJECTION FLOW RATE
C   RWST LEVEL (IN PERCENT)
C
     WRITE(*,*)ESC//"[19;5H",ESC//"[34;40m",
CIBM  WRITE(*,*)ESC//"[19;4H",ESC//"[34;40m",
    1ULBR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
    1HORZBAR//HORZBAR//CUBR//HORZBAR//HORZBAR//
    1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//URBR
     WRITE(*,*)ESC//"[18;13H",VERTBAR
     WRITE(*,*)ESC//"[17;13H",VERTBAR
     WRITE(*,*)ESC//"[16;13H",CRBR//
    1HORZBAR//HORZBAR//HORZBAR//
    1HORZBAR//HORZBAR//
    1HORZBAR//HORZBAR//HORZBAR//
    1HORZBAR//HORZBAR
     WRITE(*,*)ESC//"[20;5H",
    1VERTBAR
     WRITE(*,*)ESC//"[20;22H",
    1VERTBAR
     WRITE(*,*)ESC//"[21;5H",
    1VERTBAR
     WRITE(*,*)ESC//"[21;22H",
    1VERTBAR
     WRITE(*,*)ESC//"[22;5H",
    1VERTBAR
     WRITE(*,*)ESC//"[22;22H",
    1VERTBAR
     WRITE(*,*)ESC//"[23;5H",
    1LLBR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
    1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
    1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//LRBR
     WRITE(*,*)ESC//"[20;7H",ESC//"[37;40m",
    1"SAFETY INJ."
     WRITE(*,*)ESC//"[21;7H",
    1"FLOW"
     WRITE(*,11)ESC//"[21;13H",((WESFDC+WESFCL+WESFHL)*2.2)
     WRITE(*,*)ESC//"[22;7H",
    1"RWST"
     XCRLS=(ARWST*1000.0*ZWRWST)/MRWST0
     IF(XCRLS.GT.1.0)XCRLS=1.0
     WRITE(*,11)ESC//"[22;13H",XCRLS
C
C   PRINTOUT STEAM GENERATOR BLOCK
C
C    THIS ROUTINE PRINTS OUT
C
C    STEAM GENERATOR LEVEL
C
     WRITE(*,*)ESC//"[37;40m"
     WRITE(*,*)ESC//"[19;28H",ESC//"[31;40m",
CIBM  WRITE(*,*)ESC//"[19;27H",ESC//"[31;40m",
```

```
     1ULBR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//CUBR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//URBR
      WRITE(*,*)ESC//"[20;28H",
     1VERTBAR
      WRITE(*,*)ESC//"[20;47H",
     1VERTBAR
      WRITE(*,*)ESC//"[21;28H",
     1VERTBAR
C     WRITE(*,*)ESC//"[21;47H",
C    1CRBR
      WRITE(*,*)ESC//"[22;28H",
     1VERTBAR
      WRITE(*,*)ESC//"[22;47H",
     1VERTBAR
      WRITE(*,*)ESC//"[23;28H",
     1LLBR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//LRBR
      WRITE(*,*)ESC//"[20;33H",ESC//"[37;40m",
     1"STEAM GEN."
      WRITE(*,*)ESC//"[21;30H",
     1"UNF MS"
      WRITE(*,*)ESC//"[22;30H",
     1"FLT MS"
      X=(MWUS*2.2)
      WRITE(*,11)ESC//"[21;38H",X
      X=(MWBS*2.2)
      WRITE(*,11)ESC//"[22;38H",X
C
C    PRINTOUT CONTAINMENT BLOCK
C
C    THIS ROUTINE PRINTS OUT
C
C    CONTAINMENT PRESSURE
C    CONTAINMENT HYDROGEN MASS CAVITY
C    CONTAINMENT HYDROGEN MASS UPPER COMPARTMENT
C    FISSION PRODUCT NOBLES IN CONTAINMENT
C
     WRITE(*,*)ESC//"[7;49H",ESC//"[31;40m",
CIBM  WRITE(*,*)ESC//"[7;48H",ESC//"[31;40m",
    1ULBR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
    1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
    1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
    1HORZBAR//
    1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//URBR
     WRITE(*,*)ESC//"[8;49H",
    1VERTBAR
     WRITE(*,*)ESC//"[8;73H",
    1VERTBAR
```

```
      WRITE(*,*)ESC//"[9;49H",
     1VERTBAR
      WRITE(*,*)ESC//"[9;73H",
     1VERTBAR
      WRITE(*,*)ESC//"[10;49H",
     1VERTBAR
      WRITE(*,*)ESC//"[10;73H",
     1VERTBAR
      WRITE(*,*)ESC//"[11;49H",
     1VERTBAR
      WRITE(*,*)ESC//"[11;73H",
     1VERTBAR
      WRITE(*,*)ESC//"[12;49H",
     1VERTBAR
      WRITE(*,*)ESC//"[12;73H",
     1VERTBAR
      WRITE(*,*)ESC//"[13;49H",
     1LLBR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//CDBR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//LRBR
      WRITE(*,*)ESC//"[8;53H",ESC//"[37;40m",
     1"CONTAINMENT"
      WRITE(*,*)ESC//"[9;51H",
     1"UC PRESS."
      WRITE(*,*)ESC//"[10;51H",
     1"NOB GAS"
      WRITE(*,*)ESC//"[11;51H",
     1"UC H2"
      WRITE(*,*)ESC//"[12;51H",
     1"CAV. H2"
      WRITE(*,11)ESC//"[9;63H",(PA/6804.0)
      WRITE(*,11)ESC//"[10;63H",((QGFPA+QGFPB+
     1QGFPC+QGFPD+QGFPI+QGFPU)*2.2)
      WRITE(*,11)ESC//"[11;63H",MH2A*2.2
      WRITE(*,11)ESC//"[12;63H",MH2C*2.2
C     WRITE(*,*)ESC//"[18;38H",CDBR
C
C     PRINTOUT FEEDWATER FLOW BLOCK
C
C     THIS ROUTINE PRINTS OUT
C
C     FEEDWATER FLOW TO THE STEAM GENERATORS
C
      WRITE(*,*)ESC//"[19;50H",ESC//"[31;40m",
CIBM  WRITE(*,*)ESC//"[19;49H",ESC//"[31;40m",
     1ULBR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//URBR
      WRITE(*,*)ESC//"[21;47H",CRBR//HORZBAR//HORZBAR//CLBR
```

```
      WRITE(*,*)ESC//"[20;50H",
     1VERTBAR
      WRITE(*,*)ESC//"[20;71H",
     1VERTBAR
C     WRITE(*,*)ESC//"[21;50H",
C     1CLBR
      WRITE(*,*)ESC//"[21;71H",
     1VERTBAR
      WRITE(*,*)ESC//"[22;50H",
     1VERTBAR
      WRITE(*,*)ESC//"[22;71H",
     1VERTBAR
      WRITE(*,*)ESC//"[23;50H",
     1LLBR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//
     1HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//HORZBAR//LRBR
      WRITE(*,*)ESC//"[20;53H",ESC//"[37;40m",
     1"FEEDWATER"
      WRITE(*,*)ESC//"[21;52H",
     1"UNF FLW"
      WRITE(*,*)ESC//"[22;52H",
     1"FLT FLW"
      WRITE(*,*)ESC//"[37;40m"
      X=WWFWUS*2.2
      WRITE(*,11)ESC//"[21;62H",X
      X=WWFWBS*2.2
      WRITE(*,11)ESC//"[22;62H",X
C     ROUTINE TO PRINT OUT STATUS OF CERTAIN PARAMETERS
C
C     PARAMETERS ADDRESSED ARE:
C     SAFETY INJECTION: HPI(5), LPI(6)
C     RELIEF VALVES: SRV(42), PORVS(44)
C     NORMAL FLOWS: AFW(154), CHARGING (11), MFW(157)
C     REACTOR VESSEL: FAILED(3), CORE UNCOVERY(49)
C     CONTAINMENT: WATER IN CAVITY(57), SPRAYS(103)
C     HYDROGEN BURN: LOWER CMPT(75), UPPER CMPT(102)
C     RECIRCULATION(181), CST SOURCE(191)
C     CONTAINMENT INTEGRITY COMPT A(119,120) COMPT D(129,130)
C
C     HIGH PRESSURE INJECTION FLAG
     IF(IEVNT(5).EQ.1)WRITE(*,*)ESC//"[34;43m"
     1,ESC//"[1;60H","HPI"
      IF(IEVNT(5).EQ.0)WRITE(*,*)ESC//"[34;40m"
     1,ESC//"[1;60H","HPI"
C     LOW PRESSURE INJECTION FLAG
     IF(IEVNT(6).EQ.1)WRITE(*,*)ESC//"[34;43m"
     1,ESC//"[1;65H","LPI"
      IF(IEVNT(6).EQ.0)WRITE(*,*)ESC//"[34;40m"
     1,ESC//"[1;65H","LPI"
C     SAFETY RELIEF FLAG
     IF(IEVNT(42).EQ.1)WRITE(*,*)ESC//"[34;43m"
```

```
     1,ESC//"[1;70H","SRV"
      IF(IEVNT(42).EQ.0)WRITE(*,*)ESC//"[34;40m"
     1,ESC//"[1;70H","SRV"
C     PRESSURIZER PORV
      IF(IEVNT(44).EQ.1)WRITE(*,*)ESC//"[34;43m"
     1,ESC//"[1;75H","PORV"
      IF(IEVNT(44).EQ.0)WRITE(*,*)ESC//"[34;40m"
     1,ESC//"[1;75H","PORV"
C     AUX FEED WATER
      IF(IEVNT(154).EQ.1)WRITE(*,*)ESC//"[34;43m"
     1,ESC//"[2;60H","AFW"
      IF(IEVNT(154).EQ.0)WRITE(*,*)ESC//"[34;40m"
     1,ESC//"[2;60H","AFW"
C     MAIN FEED WATER
      IF(IEVNT(157).EQ.0)WRITE(*,*)ESC//"[34;43m"
     1,ESC//"[2;65H","MFW"
      IF(IEVNT(157).EQ.1)WRITE(*,*)ESC//"[34;40m"
     1,ESC//"[2;65H","MFW"
C     CHARGING FLOW
      IF(IEVNT(11).EQ.1)WRITE(*,*)ESC//"[34;43m"
     1,ESC//"[2;70H","CVCS"
      IF(IEVNT(11).EQ.0)WRITE(*,*)ESC//"[34;40m"
     1,ESC//"[2;70H","CVCS"
C     REACTOR VESSEL FAILED FLAG
      IF(IEVNT(3).EQ.1)WRITE(*,*)ESC//"[34;43m"
     1,ESC//"[3;60H","RVF"
      IF(IEVNT(3).EQ.0)WRITE(*,*)ESC//"[34;40m"
     1,ESC//"[3;60H","RVF"
C     CORE UNCOVERED FLAG
      IF(IEVNT(49).EQ.1)WRITE(*,*)ESC//"[34;43m"
     1,ESC//"[3;65H","CUC"
      IF(IEVNT(49).EQ.0)WRITE(*,*)ESC//"[34;40m"
     1,ESC//"[3;65H","CUC"
C     WATER IN CAVITY FLAG
      IF(IEVNT(57).EQ.1)WRITE(*,*)ESC//"[34;43m"
     1,ESC//"[3;70H","CWET"
      IF(IEVNT(57).EQ.0)WRITE(*,*)ESC//"[34;40m"
     1,ESC//"[3;70H","CWET"
C     CONTAINMENT SPRAYS FLAG
      IF(IEVNT(103).EQ.1)WRITE(*,*)ESC//"[34;43m"
     1,ESC//"[3;75H","CSS"
      IF(IEVNT(103).EQ.0)WRITE(*,*)ESC//"[34;40m"
     1,ESC//"[3;75H","CSS"
C     HYDROGEN BURN LOWER COMPARTMENT
      IF(IEVNT(75).EQ.1)WRITE(*,*)ESC//"[34;43m"
     1,ESC//"[4;60H","BURN LC"
      IF(IEVNT(75).EQ.0)WRITE(*,*)ESC//"[34;40m"
     1,ESC//"[4;60H","BURN LC"
C     HYDROGEN BURN UPPER COMPARTMENT
      IF(IEVNT(102).EQ.1)WRITE(*,*)ESC//"[34;43m"
     1,ESC//"[4;70H","BURN UC"
      IF(IEVNT(102).EQ.0)WRITE(*,*)ESC//"[34;40m"
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```
     1,ESC//"[4;70H","BURN UC"
      WRITE(*,*)ESC//"[37;40m"
C     RECIRCULATION IS BEING USED
      IF(IEVNT(181).EQ.1)WRITE(*,*)ESC//"[34;43m"
     1,ESC//"[5;60H","RECIRC"
      IF(IEVNT(181).EQ.0)WRITE(*,*)ESC//"[34;40m"
     1,ESC//"[5;60H"," RWST "
C     CONDENSATE STORAGE TANK BEING USED
      IF(IEVNT(191).EQ.1)WRITE(*,*)ESC//"[34;43m"
     1,ESC//"[5;70H","CST EMPTY"
      IF(IEVNT(191).EQ.0)WRITE(*,*)ESC//"[34;40m"
     1,ESC//"[5;70H","CST AVAIL"
      WRITE(*,*)ESC//"[37;40m"
C     CONTAINMENT FAILED IN COMPT A
      IF((IEVNT(119)+IEVNT(120)).GT.0)WRITE(*,*)ESC//"[34;43m"
     1,ESC//"[6;60H","UC FAILED"
      IF((IEVNT(119)+IEVNT(120)).EQ.0)WRITE(*,*)ESC//"[34;40m"
     1,ESC//"[6;60H","UC ISOLT"
C     CONTAINMENT FAILED IN COMPT D
      IF((IEVNT(129)+IEVNT(130)).GT.0)WRITE(*,*)ESC//"[34;43m"
     1,ESC//"[6;70H","LC FAILED"
      IF((IEVNT(129)+IEVNT(130)).EQ.0)WRITE(*,*)ESC//"[34;40m"
     1,ESC//"[6;70H","LC ISOLT"
      WRITE(*,*)ESC//"[33;40m"
     1,ESC//"[24;65H","RLS-PCSI2.1"
C     THIS LINE ADDS TITLE LINE TO
C     THE BOTTOM OF THE SCREEN
C     THE FIRST 60 CHARACTERS ARE PRINTED
      WRITE(*,*)ESC//"[37;40m"
     1,ESC//"[24;1H"
      WRITE(*,3000)(ITITL(I,1),I=1,60)
3000  FORMAT(1X,60A1)
      RETURN
      END
```

VITA

Ricky Lynn Summitt was born in Loudon, Tennessee on April 25, 1958. He graduated from Loudon High School in June 1976. He entered the University of Tennessee, Knoxville the following September and received his Bachelor of Science in Nuclear Engineering in December 1980. He accepted a position with Duke Power Company in Charlotte, North Carolina after graduation. He was a lead engineer for the McGuire Nuclear Station and Catawba Nuclear Station full-scope probablistic risk assessments. He served as chairman of the Electric Power Research Institute software user's group and was a member of the Atomic Industry Forum's probabilistic risk assessment committee. In February 1986 Mr. Summitt accepted a position with International Technology (IT) Corporation as the manager of probabilistic risk assessment. During his time at IT Corporation he was appointed project manager for probabilistic risk assessment and severe accident issue resolution for the USDOE-sponsored Advanced Reactor Severe Accident Program. He received his professional engineering license from North Carolina in 1986 and is also licensed in Tennessee and South Carolina. In May 1989 Mr. Summitt joined SAROS, Inc. as a junior partner. While at SAROS he was project and technical manager for Carolina Power and Light Company's H. B. Robinson and Shearon Harris Nuclear Plant Individual Plant Examinations. In 1988 he entered the University of Tennessee, Knoxville to begin work toward a Master of Science in Nuclear Engineering which he received in May, 1994.