KAERI/AR-524/99

액체금속로의 확률론적 위해도 평가방법 검토

(Review of PRA Methodology for LMFBR)

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본 보고서를 1998년도 "안전해석기술개발"과제의 기술현황분석 보고서로 제출합니다.

1999 년　　2 월

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요 약 문

확률론적 위해도 평가방법 (PRA, Probabilistic Risk Assessment)은 설계 또는 운전 중인 원자력 발전소의 안전성을 종합적으로 파악하기 위한 방법으로 근래에 들어 PRA의 수행은 많은 발전소의 인허가 요건이 되고 있다. 한국 원자력연구소에서는 액체금속로인 KALIMER를 개발하고 있으며, 세계적인 인허가 요건에 따라 추후에 KALIMER의 PRA도 수행하여야 할 것으로 예상된다. 기존의 PRA 방법은 경수로나 중수로를 중심으로 정립이 되어 있으나, 액체금속로의 경우에 있어서는 그 설계가 기존 경/중수로와 전혀 상이하며 사고 진행도 다르므로 액체금속로를 위한 새로운 PRA 방법이 필요하다. KALIMER의 기본 설계가 GE에서 개발한 PRISM 발전소와 유사하므로 본 보고서에서는 PRISM 발전소의 PRA 방법론을 정리하고 이 방법론의 KALIMER PRA에 대한 적용 가능성에 대하여 살펴본다. 기존의 기존 경/중수로에 대한 PRA는 주로 발전소의 안전성 평가에 그 초점이 맞추어져 있었으나, 미국 NRC는 액체금속로에 대하여서는 PRA를 이용하여 단지 발전소의 안전성을 평가할 뿐만이 아니라 PRA 결과를 이용하여 설계 개선안도 도출하기를 요구하였다. 이와 같은 요구를 만족시키기 위하여 GE에서는 PRISM 발전소에 대한 PRA를 수행하였다. PRISM PRA 방법은 기존의 경수로 PRA 방법과는 많은 차이점을 보이고 있다. 이는 노형의 물리적 차이로 인하여 초기사건, 사고 진행 과정 및 방사능 물질의 누출과정이 기존 경/중수로와 전혀 다른 양상을 보여주고 있기 때문이다. PRISM PRA 방법은 기본적으로 다음의 5가지 과정으로 이루어져 있다: (1) 초기 사건의 도출, (2) 계통 사건 수목의 구축, (3) 노심 대응 사건 수목의 구축, (4) 격납용기 대응 사건 수목의 구축 및 (5) 방출 영향과 위해도 평가. PRISM PRA의 결과에 따르면 PRISM 발전소의 위해도는 기존 경/중수로에 비하여 매우 적은 것으로 나타나고 있다. 본 보고서에서는 PRISM PRA의 각 과정에 대해 경/중수로 PRA 방법론과 비교를 하여 그 차이점을 살펴보았으며, 또한 KALIMER PRA를 수행하기 위하여 필요한 작업들을 정리하였다. 아울러, KALIMER PRA를 위하여 필요한 컴퓨터 코드 및 자료도 정리하였다.

SUMMARY

Probabilistic Risk Assessment (PRA) has been widely used as a tool to evaluate the safety of NPPs (Nuclear Power Plants), which are in the design stage as well as in operation. Recently, PRA becomes one of the licensing requirements for many existing and new NPPs. KALIMER is a Liquid Metal Fast Breeder Reactor (LMFBR) being developed by KAERI. Since the design concept of KAIMER is similar to that of the PRISM plant developed by GE, it would be appropriate to review the PRA methodology of PRISM as the first step of KALIMER PRA. Hence, in this report, we review the PRA methodology of the PRISM plant. One of the design requirements of the PRISM plant is to apply PRA to the design process. Specifically, it is required that: (1) PRA techniques shall be applied to the design process to ensure public health and safety risk, including that due to beyond design basis accidents (BDBA) is acceptably low; and (2) numerical risk limits shall be used to guide judgment of the design adequacy with respect to public risk. The methodology of PRA depends upon the design of NPP. Since the LMFBR has a totally different design from the Light Waters Reactor (LWR) and/or Pressurized Heavy Water Reactor (PHWR), the new PRA methodology is required for PRA of LMFBR like PRISM and/or KALIMER. The PRA methodology of the PRISM plant consists of the following five major tasks: (1) development of an initiating event list, (2) development of System Event Trees, (3) development of Core Response Event Trees, (4) development of Containment Response Event Trees, and (5) Consequences and risk estimation. The estimated individual and societal risk measures show that the risk from a PRISM module is substantially less than the NRC goal. In the report, each task of PRISM PRA is compared to the corresponding part of LWR/PHWR PSA performed in Korea. The computer codes and data used in PRISM PRA are also compared to the computer codes and data available in Korea. Based on the reviewed results, the required works for KALIMER PRA are identified and summarized.

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1. INTRODUCTION

Probabilistic Risk Assessment (PRA) models developed for Nuclear Power Plants (NPPs) provide valuable information and insight that can make important contributions to the process of evaluating safety issues [1]. So, PRA has been widely used as a tool to evaluate the safety of NPPs, which are in the design stage as well as in operation. In general, PRA is classified into three levels. In Level 1 Probabilistic Safety Assessment (PSA), the responses of mitigating systems to initiating events are assessed. As a result of assessment, the core damage frequency (CDF) is estimated. Level 2 PSA analyzes the containment response under accidents. Finally, the consequences and risk after containment failures such as acute and latent fatalities are assessed in Level 3 PSA. Generally, we use the term, PRA when we indicate whole Level 1,2 and 3 PSA.

The results of PRA provide us with insights on the weak points of NPP design and/or operation. So, we can improve the design and/or operation of NPP based on the results of PRA. In most countries, the PRA of NPPs is required as a part of the licensing process.

Even though, the basic concept of PRA is the same for all kinds of reactors, the methodology of PRA depends upon the design of NPPs. Up to now, the PRA methodology has been developed mainly for the Light Water Reactor (LWR), and/or Pressurized Heavy Water Reactor (PHWR). KALIMER is a Liquid Metal Fast Breeder Reactor (LMFBR) being developed by KAERI. Since the LMFBR has a totally different design from that of the LWR and/or PHWR, the new PRA methodology is required for the LMFBR like KALIMER.

As the basic design concept of KAIMER is similar to that of the PRISM plant developed by GE, it would be appropriate to review the PRA methodology of PRISM [2, 3] as the first step of KALIMER PRA. The PRA methodology of the PRISM plant has the same goal as that of LWR/PHWR PRA. However, owing to the differences in design, the detailed procedures and data are different from those of LWR/PHWR PRA. Hence, in this report, we will review each task of PRISM PRA, and compare it with the corresponding part of LWR and/or PHWR PSA performed in Korea [4-6]. In addition, the parts that are not modeled appropriately in PRISM PRA are identified, and the recommendations for KALIMER PRA are also summarized.

Section 2 of the report provides a summary of the risk assessment model and quantification procedures of PRISM PRA. This section also discusses the particular characteristics of the PRISM plant, which have a significant effect on the risk model structure and results.

Section 3 contains the brief assessment of the initiating events, system response event trees, core response event trees, vessel and containment response event trees, institutional decisions, and public consequences of PRISM PRA. The rationale for defining the events in the risk model, procedures, data, data sources, and results of assessing the probabilities of these events are discussed briefly in this section. Each task of PRISM PRA is compared to the corresponding part of LWR/PHWR PSA. This section also presents the risk results and compares them to the NRC safety goals.

Finally, in Section 4, the recommendation for KALIMER PRA are presented based on the reviewed results. The computer codes and data used in the PRISM PRA are also compared to the computer codes and data available in Korea.

2. APPROACH OVERVIEW OF PRISM PRA

This section summarizes the PRISM PRA approach and the quantification procedures and databases of PRISM PRA. In Section 3, the procedures and actual data used in PRISM PRA are discussed in detail.

One of the design requirements of the PRISM plant is to apply PRA to the design process. Specifically, it is required that:

1. PRA techniques shall be applied to the design process to ensure the public health and safety risk, including that due to beyond design basis accidents (BDBA) is acceptably low; and
2. Numerical risk limits shall be used to guide judgment of the design adequacy with respect to public risk.

The numerical measures of risk adopted to carry out these requirements have been derived from the NRC safety goal policy statement [7]. The risk measures are given in Table 1.

Consistent with the PRISM design requirements and the intent of the above NRC policy statement, this preliminary probabilistic risk assessment has been conducted with the following objectives [2]:

1. To evaluate the extent to which the PRISM power plant meets the quantitative goals of Table 1.
2. To delineate system relationships which must be understood for risk management. This includes :
   - Identifying major contributors to risk.
   - Estimating the sensitivity of the risk to uncertainty in input data.
   - Characterizing the radioactivity release patterns to asses potential for post-accident risk management.

It should be noted that this PRA analysis is a part of an iterative process involving the interaction between design and PRA activities. It is the objective of PRA activities to seek accurate estimates of probabilities and consequences. However, conservative assumptions had to be used in this analysis when necessary to expedite the feedback of PRA to the design and data base could very well show that the results of PRA are unduly conservative.

The PRISM power plant analyzed in the PRA is the reference (metal-core) design described in the main body of this PSID [2]. The plant has been assumed to be located on a GESSAR-II site. The study has been confined to the following scope :

1. The study does not include risks from acts of sabotage or normal plant effluent releases.

2. The study has been confined to accidents in a single module. The module affected is assumed to be operating at full power when an accident is assumed to occur. In particular, the study has not considered startup accidents, partial power operation, or situations where one module in the same power block is out for refueling or for other reasons.

3. The study has been confined to core-related accidents. In particular, accidents related to radioactivity sources outside the reactor vessel, e.g., radwaste systems, have been exclude. However, the radioactivity sources in the PRA include :

- driver fuel;
- inner and radial blankets;
- activated primary sodium;
- spent fuel stored in-vessel.

To simplify the study, end of equilibrium cycle radioactivity inventory has been conservatively assumed at the time of accident.

This PRA has been developed in accordance with the following guidelines:

1. To use state-of-the-art methods and data.
2. To use the mean values as estimates for the risk measures.
3. To incorporate uncertainties in important phenomena, key assumptions, and input data in the risk assessment.

The risk from operating a NPP results from sequences of events which lead to the release of radioactive material to the environment. The definition of these events and the extent to which each event is analyzed could significantly affect the accuracy of the risk results. In principle, it is desirable to use a fine classification of these events if the risk contribution is significant or uncertainty in the risk contribution is large. Conversely, events which have insignificant impact on the risk or lead to comparable risk contribution may be grouped without much loss of accuracy. The net result of applying these principles is a risk model which highlights major risk contributors with minimum uncertainties introduced by inadequate event definitions.

Section 2.1 presents the specific characteristics of the PRISM power plant which have been considered for event definition in the risk model. Section 2.2 provides a summary of the risk model structure. Section 2.3 presents the procedures and data sources used for quantifying the risk.

2.1 NATURE OF THE PRISM RISK

The PRISM power plant has distinctive features for preventing accidents, for limiting the extent and speed of accident progression should an accident occur, and for retaining the fission products. As discussed below, the effect of these features on the probability and consequences of accidents has been to reduce the relative significance of independent failures and slow transients. Consequently, the risk model developed for this PRA highlights dependent, concurrent, and coherent failures.

2.1.1 ACCIDENT PREVENTION

The PRISM design places strong emphasis on the reliability principles of redundancy (and diversity), testing, use of passive concepts for power control and heat removal, and fail-safe or self-correcting failure provisions. The following examples illustrate this emphasis :

- Redundancy and Diversity

The PRISM Reactor Shutdown System (RSS) uses six control rods, although one rod is adequate to shut down the reactor. Another example of utilizing redundancy is the use of quadruply redundant channels for process data handling and transmission. The application of diversity is illustrated by the use of in-vessel instruments which measure different process parameters, are placed in different locations, and are exposed to different environments.

- Testing

The PRISM reactor uses continuous monitoring of its Reactor Protection Systems (RPS) channels. Continuous monitoring is also used at the interface of the RPS and other systems, e.g., Plant Control System (PCS). The PRISM reactor also uses frequent testing by operation of some of the critical components, such as the control rod drive motors and control rods.

- Use of Passive Concepts for Power Control and Heat Removal

An example of the first concept is the thermal expansion of the control rod drive lines and core load pads in response to increase in primary coolant temperature. The reactivity resulting from this expansion would offset the positive reactivity resulting from coolant density changes caused by coolant heating. This results in a net negative temperature coefficient of reactivity for the primary coolant temperature. An example of the second concept is the Reactor Vessel Auxiliary cooling System (RVACS) which removes decay heat by natural convection.

- Fail-Safe and Self-Correcting Failure Provisions

These provisions utilize two different principles to respond to a failure.

1. Transfer to a more reliable or at least as reliable configuration.

   An example of applying this principle is the fault-tolerant quad-redundant logic used in the PRISM RPS. The system uses a 2-out-of-3 logic when all channels are operable (one channel on rotating standby), transfers to the equally reliable configuration of 2-out-of-3 logic when one channel fails (the failed channel is excluded until repaired), and transfers to the more reliable configuration of 1-out-of-2 logic when two channels fail (the failed channels are excluded until repaired).

2. Safe transfer to desired state with appropriate use of stored energy.

   An example of applying this principle is the use of stored energy to control flow coast down.

The above characteristics have resulted in high reliabilities of the PRISM systems. In particular, the probability of independent failures under nominal design conditions has been estimated to be extremely low. This raises the relative importance of dependent failures and of operating conditions outside the design envelope as risk contributors.

2.1.2 LIMITING EXTENT AND SPEED OF ACCIDENT PROGRESSION

Threshold phenomena such as fuel melting, coolant boiling, clad rupture and structural failure could significantly impact the course of an accident and the resulting consequences. The phenomena may result from excessive energy generation or inappropriate energy distribution between different parts of the system. In particular, when the rate of energy transfer into a part of the system exceeds the rate at which it is transferred out, the deposited energy may result in melting, boiling, creep rupture, etc. Two types of system time parameters characterize these rates: 1) the reactor period, or equivalently the net reactivity rate and magnitude, which characterizes the rate of nuclear energy generation, and 2) the time constant of the fuel, clad, core, coolant, etc., which characterizes the rate of heat transfer out.

The PRISM design uses the following provisions to maintain a long reactor period (slow rate of energy generation) :

1. small reactivity additions (~20 ¢) if a control rod is inadvertently withdrawn ;
2. inherent negative reactivity feedback if the fuel or the primary coolant temperature increases.

The reference PRISM core uses a I-Pu-Zr metal fuel with HT9 cladding. The core time constant is characteristically short (~0.3 sec or ~10% of a typical oxide core). This results in a fast heat transfer from the fuel to the coolant, or equivalently, the deposition of only a small fraction of the heat in the fuel with a corresponding low fuel temperature. Despite this advantage, the possibility of threshold phenomena cannot be excluded due to the following conditions:

1. relatively low melting point of the metal fuel (~1150℃)
2. formation of fuel/clad eutectic alloy which may begin at ~725℃
3. primary sodium boiling (~880℃) or voiding due to the fast rate of energy transfer in.

The threshold temperatures shown above indicate the potential vulnerability of cladding being the first to fail as a result of eutectic formation. This is different from typical accident scenarios in LMRS using oxide fuels. In such cases, unprotected loss of flow accidents (LOF) lead to sodium voiding, followed by clad melting, then fuel melting. On the other hand, unprotected transient over power (TOP) accidents lead to the opposite order of fuel melting, then clad melting.

Sweep out of the molten clad from the core in the above oxide-core scenarios results in positive reactivity additions. On the other hand, fuel/clad eutectic melt sweep out of the metal core could lead to negative reactivity additions and shutdown due to fuel removal from the core region. In this regard, then, the eutectic formation can be viewed as another mechanism for limiting the extent of accident progression.

The above characteristics reduce the relative importance of slow transients as public risk contributors, or equivalently, highlight the relative importance of fast transients involving rapid reactivity additions or rapid loss of flow as potential risk contributors.

2.1.3 RADIOACTIVE MATERIAL RETENTION

The reference PRISM metal fuel has the following characteristics which could significantly affect the timing and mix of radioactive material releases in case of an accident.

1. For burnups greater than 2 at %, the fuel is virtually transparent to fission gas, i.e., the fission gas generated by fission is transmitted to the fission gas plenum at the rate of generation.

2. Accidents may be terminated at relatively low temperatures (<1150℃) involving molten fuel/clad eutectic alloy or molten fuel. The low temperature presents a strong potential for retention of strontium (Sr) and tellurium (Te) isotopes (which may be major risk contributors in other fuels) in the fuel body. Therefore, if the fuel is retained inside the vessel, these radionuclides will not be released.

These considerations have been reflected in the definitions of accident scenarios and release categories of the risk model described below.

2.2 RISK MODEL

The risk analysis of a PRISM module starts with the initial condition that the module is operating at full power. An event for which the reactor would be or should be shut down is assumed to occur. Such an event is called an initiating event (IE). In response to the initiating event, the module is expected to control the nuclear power generation, coolant flow, and heat removal processes to bring the reactor to a safe shutdown until the cause of shutdown is removed. In the course of this transition to shutdown or following the nuclear shutdown, imbalance between these processes may occur. If such imbalances do not cause clad or core damage, the module resumes operation after the cause of shutdown is removed.

Otherwise, the situation is termed an "accident", and the module is assumed not to resume operation.

The risk model defines the events, event sequences or scenarios, and the statistical relationships and dependencies between them, which are required for estimating the probabilities and consequences. An overview of the risk model structure is delineated in Figure 1. The structure contains the following major elements:

- Initiating Events:

  Twenty-one mutually exclusive and collectively exhaustive events have been defined in the risk model. These include normal shutdown for refueling, spurious shutdown signal, forced shutdown, malfunctions leading to three ranges of reactivity additions, partial or complete loss of forced flow, partial or complete loss of heat removal capability, station blackout, partial core blockage, core support and vessel failures, and three levels of earthquake events.

- System Event Sequences and Accident Types :

  For each initiating event, a system event tree is developed to identify possible sequences which lead to safe shutdown and those which lead to accidents. The event trees include response of the power control systems (plant control system [PCS], reactor protection system [RPS], reactor shutdown system [RSS] and inherent reactivity feedback features), flow control systems (pump trip and flow coast down system), and heat removal systems (via balance of plant [BOP], intermediate heat exchanger [IHX], or RVACS). The sequences are formed from possible combinations of success and failure of the various systems. Each sequence ends either with a safe shutdown condition or one of 23 accident types. The accident types cover loss of the shutdown heat removal system after neutronic shutdown, four levels of severity of transient overpower without scram, two levels of severity of loss of flow without scram, two levels of severity of loss of heat sink without scram, two levels of severity of loss of flow without scram, two levels of severity of loss of heat sink without scram, and combinations of the above.

- Core response Event Trees and Core Damage Categories.

  For each accident type, a core response event tree is developed to identify the possible core scenarios until neutronic shutdown is accomplished. The event trees include reactivity feedback mechanisms for intact fuel (Doppler, thermal expansion), molten fuel and clad motion (with and without eutectic formation), and sodium voiding. The mechanisms may be adequate to cause neutronic shutdown with no further damage.

  One the other extreme, they may enhance energetic events before shutdown. Each scenario formed from possible combinations of reactivity feedback and energetic events leads to one of 12 core damage categories. The categories cover the spectrum of possible fractions and types of fission products released from the core, damage to the vessel seal as a result of accident energetics, location and coolability of fuel debris formed, if any, coolant enthalpy at the time of neutronic shutdown, and the states of the shutdown heat removal system.

- Containment Response Event Trees and Radionuclide Release Categories

  For each core damage category, a containment response event tree is developed to identify possible radioactive material transport scenarios until a stable configuration is reached. The event trees include events relevant to long term coolability, timing of vessel failure, delayed energetics resulting from potential recriticality, and radioactive material release timing and paths. Each scenario results either in complete retention of radionuclides within the reactor vessel indefinitely, or in one of nine categories of radionuclide release to the environment. The release categories are characterized by the fractions of different groups of radionuclides released as a function of time, from the incipience of the accident until a stable end state is reached.

- Institutional Response and Consequence Types

  For each radionuclide release category, the likely institutional responses in terms of the timing for evacuation and evacuation effectiveness are determined. Four types of consequences are then evaluated for each category of radionuclide release; latent and early fatalities given the above institutional responses, and latent and early fatalities assuming complete institutional failure.

The risk model combines the above elements probabilistically with proper accounting for dependencies between the events and event sequences. With the use of proper probability and consequence values, the model produces the following risk measures for one year of reactor operation: 1) probability of early fatality to an individual within a one mile radius from site, and 2) probability of latent fatality to an individual within a 10 mile radius from site. The procedures used for quantifying the probabilities and consequences are summarized below.

2.3 QUANTIFICATION PROCEDURES

A brief discussion is presented below. The detailed procedures, data sources, and data values used are contained in Section 3.

- Initiating Events

  The list of initiating events for FY85 PRISM design has been updated to reflect changes in the design. The expected frequencies of normal shutdown for refueling and forced shutdown have been updated to reflect new operation ground rules. The expected rates of reactivity faults and component failures have been updated by incorporating recently-developed reliability analyses. Data sources for the probability estimates include Nuclear

Plant Reliability Data System (NPRDS), Clinch River Plant Risk Assessment (CRPRA), and GESSAR site seismic frequency curves.

- System Event Trees

The conditional probabilities of failure of the systems in these trees, given each initiating event, have been estimated using fault trees, reliability block diagrams, the FRANCALC1 computer code, and dependency analysis. The estimates are based on appropriate component failure modes, testing, and repair. Dependency analyses used include analysis of the functional dependence of a component on its interfacing components and environment, fragility analysis of components under seismic events, and use of Beta factors which express the conditional probability of multiple component failures given that one component has been found in a failed state. Data sources used in the above analysis include NPRDS, CRPRA, the Reactor Safety Study (WASH-1400), generic fragility data, and engineering judgement.

- Core Response Event Trees

The conditional probabilities of events in these trees, given each accident type, have been analyzed by judgment based on ANL analysis using the SASSYS computer code, GE analysis using the ARIES code, and bounding analysis for reactivity feedback assessments. Fuel clad eutectic formation and fission product release during normal operation have been based on test results reported by ANL. The types and amounts of fission products released in-vessel under accident conditions have been estimated by adjusting the release fractions used in the Reactor Safety Study to account for the low temperatures of eutectic formation and fuel melting in the PRISM metal core.

- Containment Response Event Trees

The conditional probabilities of events in these trees, given each core damage category, have been estimated based on ANL assessment of fuel debris coolability and late energetics due to recriticality, on the reliability of the shutdown heat removal system to continue operation until accident conditions are removed, and on bounding calculations for the reactor vessel creep rupture under accident conditions. The cumulative fractions of the core radioactive material inventory released as a function of time has been estimated by HEDL for each accident scenario using a thermal analysis computer code. The thermal model used accounts for decay heat, sodium concrete reaction, and sodium fire in air when these conditions are present. Rates of energy generation, leakage, and attenuation of radioactive materials were assigned by judgment based on test results and analysis of similar accident situations.

- Consequences

Early and late fatalities, given each release category, have been calculated by using the MACCS computer code. Cases with and without evacuation were run. Sensitivity analyses to evaluate the importance of fuel/concrete reactions and accident mitigation were also investigated. All calculations used NUREG-1150 assumptions, e.g., shielding factors and relocation criteria. Fifty-four radioisotopes were used for the analysis. The inventory of each isotope was estimated by the ORIGEN computer code for the FY86 PRISM reference metal core. The population distribution and meteorological data for the GESSAR site were used.

- Risk Estimation

The RISKSP computer code was used to estimate the risk measures defined in the NRC safety goal policy statement, cumulative probability distributions of early and latent fatalities, and probabilities of the 23 accident types, 12 core damage categories, and nine release categories. The code used the event trees of the risk model and the probabilities described above. Uncertainties in the input probability estimates were assigned by judgment. The code propagated the uncertainties in input data using a Monte Carlo sampling procedure which appropriately accounts for statistical dependencies.

Table 1. Safety Goals

Safety Measure	Goal Safety Measure Must Not Exceed
Individual Risk: Probability of prompt fatality (per one year of a nuclear plant operation) for an average individual residing within one mile from the plant site boundary.	5×10^{-7}
Societal Risk: Probability of cancer fatality (per one year of nuclear plant operation) for population residing within 10 miles of the plant site.	1.9×10^{-6}

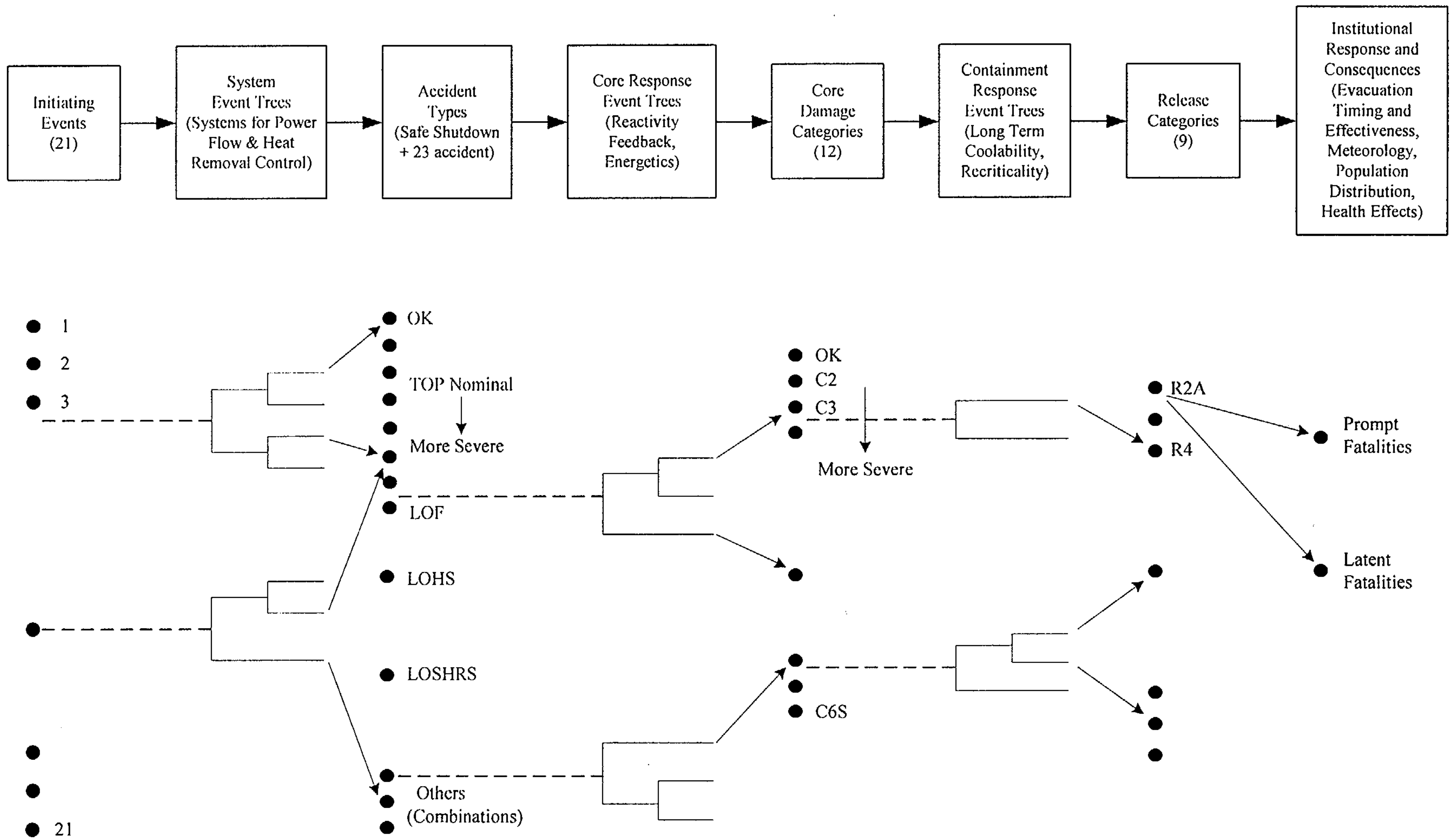

Figure 1 An Overview of The Risk Model Structure

3. RIVIEW OF PRA METHODOLOGY

This section contains a brief assessment of the initiating events, system response event trees, core response event trees, vessel and containment response event trees, institutional decisions, and public consequences of PRISM PRA. The rationale for defining the events in the risk model, procedures, data, data sources, and results of assessing the probabilities of these events are also discussed in this section. Each task of PRISM PRA methodology is briefly compared to the corresponding part of LWR/PHWR PSA methodology.

3.1 INITIATING EVENTS

In PRISM PRA, the initial condition assumed for this PRA is that the plant is operating at full power. Given this initial condition, an initiating event for a module is defined as an event for which the module would or should be shutdown. Shutdown will continue for a period of time until the cause of shutdown is removed. The cause of shutdown could be an abnormal condition such as an uncontrolled withdrawal of a control rod, or normal shutdown for refueling. The objectives of this section are to:

1. define an exhaustive list of initiating events appropriate for PRISM,
2. estimate the expected frequency of each initiating event, and
3. estimate the mission time of the shutdown heat removal system given each initiating event. This time is defined as the mean time to remove the cause of shutdown and return the affected module to its initial state of operation (MTTR).

Review of initiating events of past PRA applications and the PRISM design resulted in the list of Table 2. The list can be thought of as composed of three groups:

1. reactivity insertions other than seismic,
2. external events (primarily earthquakes), and
3. heat removal faults,

The basic approach to derive the list of initiating events in PRISM PRA is the same as that of LWR/PHWR. The derived list of initiating events covers most of LWR initiating events. However, this list has some special features like following:

1. the reactivity induced events and the seismic events are classified in detail, and
2. the BOP induced events are lumped as a big category like "Loss of Shutdown Heat Removal via BOP."

Such facts are due to the characteristics of PRISM plant. That is, for the second item, the PRISM plant has the capability to remove the heat via passive ways like RVACS. So the failure of heat removal is not a big concern in PRISM plant as in LWR/PHWR.

However, among external events, only the seismic events are analyzed. The other external events such as flooding and fire are to be included in the list of initiating events for the completeness.

3.2 SYSTEM EVENT SEQUENCES

As discussed in Section 3.1, an initiating event requires the module to be shut down until the cause of shutdown is removed. The systems responsible to realize shutdown are those for control of the module power, coolant flow, and heat removal. Possible responses of these systems may lead to safe shutdown and restart of the module operation as expected, or may lead to an abnormal situation which will henceforth be called an accident.

To systemically identify all possible accident types, a system response event tree has been developed for each initiating event. The system event trees are a part of the risk model and display the following important parameters:

1. Logical combinations of system responses which form accident sequences.
2. Dependencies between responses of the various systems.
3. Relation between accident sequences and the end states of either safe shutdown or one of twenty-three accident types.
4. Probabilities of various system responses and accident sequences.

An example of the developed system event trees is shown in Figure 2. The event trees of PRISM PRA shows three distinct patterns. The first pattern covers initiating events 1 through 18; except for the initiating event of earthquakes greater than 0.825g (initiating event 6). For this pattern, shutdown is initiated by RPS action. The second pattern is used for the large earthquake initiating event. In this case, the event tree explicitly includes response of the seismic isolators. The third pattern covers initiating events 19 through 21, where shutdown is initiated by PCS fast power runback.

Each of the system response event trees for initiating events 1 though 18 contains exactly seven events. Except for initiating event 6 (earthquake greater than 0.825g), each of the trees contains responses of the following systems:

1. Reactor Protection System (RPS).

   This system senses the need to shut down and initiates the proper signals for power, flow, and heat removal control.

2. Reactor Shutdown System (RSS).

   This system includes the control rods, control rod drive motors, and magnetic latches.

3. Inherent Reactivity Feedback Features.

   These include the control rods, their drive lines and their guide tubes, the core restraint system, load pads of the core assemblies, and the grid plate.

4. Primary Pumps.

   This includes the primary pumps and their power supply.

5. Pump Coast-down System.

6. Operating Power Heat Removal System (via Balance of Plant [BOP]).

7. Shutdown Heat Removal via IHX or RVACS.

The detailed failure criteria of the above systems, the probability models and data used for estimating the conditional probability of failure of each system are stated in Ref. [2].

As noted above, the system response event tree for the largest earthquakes differs from the ones above in that it contains the event "Seismic Isolation Function." This event is included only in the one tree because the possibility of failure of this system for other events has been determined to be unrealistic. Failure of this function has been calculated to result in large structural deformations. These deformations have been assumed to put the control rods out of the core thus resulting in a large transient overpower and loss of flow. Moreover, gross structural failure of the vessels may occur and core meltdown is not unlikely.

Since system response event trees for initiating events 10 through 21 present orderly PCS or manual shutdown, the only event of interest is the shutdown heat removal system capability to remove decay heat until ascent to full power operation.

The system event trees display three types of dependencies, either explicitly or implicitly.

1. Dependence on the Initiating Event: This dependence is accounted for in the definition of system success criteria (e.g., RPS sensors and set points which will result in scram, number of control rods which have to be inserted, duration for which SHRS must remove decay heat, degradation or loss of SHRS subsystems which must be assumed as a result of the initiating event).

2. Dependence Between System Responses: This dependence has influenced the order in which the system responses are displayed in the event trees. An example of this dependence is the failure of the RSS to insert its control rods if the RPS fails to sent a scram signal. Another example of system dependence is the successful shutdown by the RSS which renders the response of the inherent feedback system as irrelevant. These types of functional dependencies are represented in the event trees by different conditional probability estimates which depend on the preceding sequence of events, or by "straight through" or "don't care" lines which do not branch into success and failure branches for the dependent system response.

3. Dependencies Between Subsystems of a System: These dependencies are factored in the system reliability models by using beta factor method. The probability estimates shown on the event trees reflect these dependencies.

The system event trees are to be interpreted as follows. At each node where the tree branches, the top branch means that the event presented at the top heading of the tree has occurred. The lower branch means that the event did not occur. Sequences formed from the various events lead either to safe shutdown and restart of operation of the affected module (S1) or to one of twenty-three accident types. Each accident type is presented in the event tree by a letter symbol, (e.g., S, P, F, H, G) which refers to a generic accident group followed by a number (e.g., 1, 2, 3, 4) which refers to a level of severity of the accident type relative to its generic group. The level of severity increases with this number. For example, P3 is a more severe transient overpower than P1. The generic accident groups are:

1. Protected (i.e., reactor is shut down by RSS) loss of the shutdown heat removal system (LOSHR): represented in the event tree by the accident S3 and S5 (S1 stands for successful shutdown and shutdown heat removal).
2. Unprotected (i.e., reactor is not shut down because of RPS or RSS failure) transient overpower (TOP): represented in the event trees by the letter P.
3. Unprotected loss of flow (LOF): represented in the event trees by the letter F.
4. Unprotected loss of heat sink (ULOHS): represented in the event trees by the letter H.
5. Unprotected combined TOP/LOF or TOP/ULOHS: represented in the event trees by the letter G.

Table 3 contains the specific definitions of the accident types.

The concept of the above system event trees is similar to that of LWR/PHWR PSA. However, there is difference in defining the ending state of event trees. In LWR PSA, the core damage status is divided into only two categories: Core Damage State and Safe State. However, in PRISM PRA, the ending state of event trees represents not the core damage state but the accident type. It is similar to the system event tree of PHWR PSA in which the ending state of event trees is divided into 10 Plant Damage State (PDS) according to the severity of accidents. The 23 accident types are further analyzed to define the core damage state according to the transient phase until neutronic shutdown is accomplished. This will be explained in next section.

3.3 CORE RESPONSE EVENT TREES

Given one of the accident types defined in the previous section, the reactor core will go through a transient phase until neutronic shutdown is accomplished, either by natural processes or by human intervention. Core response event trees define the possible scenarios and end states of this transient phase. The end states are differentiated by the following parameters:

1. Configuration and physical form of the radioactive material source. The source includes the fuel, fission products and radioactive sodium.
2. Leak paths which may be opened up during the transient phase and which may lead to the release of sodium or radioactive material from the reactor vessel.
3. Primary coolant enthalpy and state of the shutdown heat removal system.

The above parameters depend on the following factors :

1. Irradiation history of the reactor core before the accident occurs.
2. Mechanisms of radioactive material release from the fuel and cladding which may be active in the course of an accident.
3. Mechanisms of energy generation and distribution which may be active in the course of an accident

This is followed by the event tree models which define the scenarios and end states for each accident type. An example of core response event trees is shown in Figure 3.

Core damage categories are defined in terms of the following four parameters.

1. Fission products: Fraction released and where released
2. Fuel: Fraction released physical form if outside cladding where released
3. Vessel and vessel seals damage, if any
4. Primary coolant temperature

The categories are labeled C1, C2, ..., C6 for end states with the shutdown heat removal system (SRHR) available and C1S through C6S for corresponding cases with SHRS unavailable. The damage severity increases as one moves from C1 to C6 (or C1S to C6S).

There are no event trees like the core response event trees in LWR/PHWR PSA which identify the possible core scenarios until neutronic shutdown is accomplished. In addition, the molten core behavior under accident conditions might be different owing to the differences in fuel materials. The reference PRISM core uses a I-Pu-Zr metal fuel. On the other hand, KALIMER core uses Na-U235-u238 metal fuel. Such differences in fuel material can result in a different melting temperature and different fission products. Again, it can cause different behavior of the core after the accidents. These aspects are to be analyzed for KALIMER PRA.

3.4 CONTAINMENT RESPONSE EVENT TREES

The purpose of the Containment Response Event Trees is to determine, given each of the 12 Core Damage Categories described previously, the probability of reaching each of the possible Containment Release Categories (including the category "OK" i.e. no release). The 12 Core Damage Categories represent a mutually exclusive, collectively exhaustive set of states of the reactor core at the end of the early phase of accident response. This early phase is termed here "core response" while the later phase is referred to as "containment response". The Containment Release Categories likewise represent a set of mutually exclusive collectively exhaustive states of the system. These categories differ from one another in the magnitude and timing of radioisotopes released, and include one category to cover the remaining possibility that there is no release. An example of containment response event trees is shown in Figure 4. The 9 containment release categories are shown in Table 4.

There are two major tasks which make up the containment response analysis phase of this risk assessment. The first task is the development of an event tree for each of the 12 Core Damage Categories. The second task is to perform a computer calculation simulating a typical event sequence leading to each of the Containment Release Categories, which are the outcomes of these trees. The result of these computer calculations is a quantitative description of the nature of the release. Specifically, the cumulative fraction of the core inventory of each of five isotope groups released as a function of time is determined. These quantitative descriptions of the release categories can then be used in the next phase of the risk assessment to calculate the expected public consequences given each release.

The concept of a containment response event tree is the same as that of LWR/PHWR PRA. However, owing to the design differences in both reactor types, the containment response event tree is built a little bit differently. In LWR/PHWR PRA, the PDS event trees model the operation of safety systems such as containment spray and hydrogen igniter. However, in PRISM PRA, the main concern of the containment response event tree is physical phenomena such as the behavior of debris, vessel failure, sodium voiding, etc., like in the containment event trees of LWR/PHWR PSA. This is due to the fact that there are no containment safety systems in the containment of the PRISM plant.

3.5 EVALUATION OF CONSEQUENCES

For each of the quantitative release descriptions obtained in Section 3.4 for the nine Containment Release Categories, several types of public risk measures were calculated using the MACCS code. These risk measures include number of prompt fatalities, number of latent cancer fatalities, probability of prompt fatality to persons within one mile of the plant, and probability of latent cancer fatality to persons within 10 miles of the plant. Input data and assumptions needed to perform these calculations are described below.

- Source Terms

The magnitude and timing of the release of each isotope group were defined in the release categories of containment response event trees. These releases are described in terms of fraction of total core inventory. The absolute core inventory of each of a list of 54 isotopes was also provided as input. This inventory was calculated by use of the ORIGEN computer code for the FY86 PRISM reference metal core.

In addition to the nine releases calculated as described in Section 3.4 four additional releases were evaluated. These releases were chosen to test the sensitivity of the results to the fact that the releases were determined under the assumption of oxide fuel, and to test the effectiveness of terminating the releases by emergency action as at Chernobyl. Thus, in addition to releases R2A through R8S, public consequences were calculated for R4AM, R4AME, R6AM, R6AME. The suffix M indicating metal fuel and E indicating a release terminated early due to emergency action. The release descriptions for the metal fuel cases R4AM, R6AM were obtained by extrapolation from the corresponding oxide releases R4A and R6A as follows: first, the overall timing of the release was accelerated by a factor of 1.5 due to the addition of energy from the oxidation of the core in a fuel-concrete-water reaction. In addition, the total amount of fuel and solids released was increased to 15% under the assumption that the fuel-concrete reaction might produce a large fraction of aerosol-sized particles capable of being carried off with the sodium.

The emergency action cases R4AME and R6AME were obtained by truncating the R4AM release at 17 hours and the R6AM release at 50 hours.

- Societal Response to Release

The first institutional response required for calculation of the consequences of release is the time at which the authorities order an evacuation. This point in time would vary depending on the sequence of events, i.e. on the Containment Release Category and is highly uncertain. For release category R2A a warning time of 1/2 hour is used. In R2A, R6S, and R8S a 25% core meltdown has occurred at time zero; however, 15 min to 1 hr is required to melt through the vessels. Hence, the authorities may delay issuing the evacuation order, so 1/2 hour is used. In the case of R3 and R4A there has been an immediate energetic expulsion at time zero; thus there is no reason to doubt the occurrence of an accident, so a warning time of 0.3 hour is used. For all other release categories there is a gradual heat up and boil-off sodium due to the loss of heat removal.

This process takes 3 to 4 days, hence there is plenty of time for authorities to reach a proper understanding of the situation and to act in time.

A rule of thumb was used that the evacuation would be ordered when the scenario had progressed 0.6 of the time to core melt. This is 60 hours in the case of R6A and R8A, and 41 hours in the case of R6U and R8U. The next issue is the response of the public to the evacuation order. This is highly uncertain. However, a fit to actual experience with evacuation due to hazardous substance releases was obtained by using 3 population subgroups: (1) 30% delay 1 hour (2) 40% delay 3 hours, and (3) 30% delay 5 hours, then evacuate at 10 mph radially away from the plant. To simplify the calculations, the 5 hour delay was conservatively used for the whole

There are no big differences in evaluating the consequences of accidents since this procedure is independent from the design of reactors. Only the differences will be the site characteristics such as the population density of the site, etc.

3.6 THE RESULTS OF PRISM PRA

The PRA methodology of PRISM is briefly reviewed in the report. Following is the summary of PRA results for PRISM.

Individual and societal risks have been evaluated using the risk model and quantification procedures summarized in above subsections. The risk model contains an exhaustive set of accident sequences which may lead to radioactive material release from a PRISM module. Cases with and without evacuation of the population around the site have been assessed.

The estimated individual and societal risk measures are presented in Table 5. The table shows that the risk from a PRISM module is substantially less than the NRC goal. Specifically,

1. The societal risk (probability of latent cancer fatality) is less than the NRC goal by a factor of 200,000 with evacuation and a factor of 146,000 without evacuation.
2. The individual risk (probability of prompt fatality) is negligible with evacuation. Without evacuation, the individual risk is less than the NRC goal by a factor 5,400.

The societal and individual risks are dominated by the following accident sequences.

1. A large earthquake (>0.825 g ground acceleration) which results in a reactivity insertion due to core compaction and relative core-control rod motion, and causes failure of the reactor shutdown system and flow coast down system. This sequence leads to an energetic core disruption and subsequent release of radioactivity. This accident sequence accounts for 48% of the societal risk and 49% of the individual risk.
2. A large earthquake (>0.825 g ground acceleration) which results in a reactivity insertion and failure of the reactor shutdown system as above, and causes in-vessel structural damage which prevents proper thermal expansion which nominally provides the inherent reactivity feedback. This sequence leads also to an energetic core disruption and subsequent release of radioactivity. This accident sequence accounts for 35% of the societal and individual risks.
3. A failure of one or two primary electromagnetic pumps accompanied by failure of the shutdown system in such a way that credit of the control rod thermal expansion cannot be relied upon as an inherent reactivity feedback mechanism. This sequence may lead to an energetic core disruption and subsequent release of radioactivity. This accident sequence accounts for 16% of the societal risk and 11% of the individual risk.

4. A large earthquake (>0.825 g ground acceleration) which causes failure of the seismic isolators and subsequent reactivity insertion, loss of the shutdown heat removal system, and loss of the reactor shutdown system. This sequence leads to an energetic core disruption and subsequent release of radioactivity. This accident sequence accounts for 4% of the individual risk but a negligible fraction of the societal risk.

The PRISM risk is of such small magnitude that it is dominated only by the residue of structural failures and severe accidents which have extremely low probability of occurrence. This is attributed to the safety philosophy of the PRISM reactor, which resulted in :

1. limited hazard potential due to the small-size reactor core, small control rod reactivity worth, and seismic isolation ;
2. highly reliable systems for control of power, flow, and heat removal, with very little reliance on active systems for safe shutdown ;
3. limited radioactivity release potential due to inherent safety characteristics and the large thermal capacity and low pressure of the primary coolant.

Table 2. Initiating Events Frequency And Mean Time To Recover

	Initiating Event (IE)	f (1)	t_m (2)
1	Reactivity Insertion 0.07-0.18	600	1.0E-4
2	Reactivity Insertion 0.18-0.36	600	1.0E-4
3	Reactivity Insertion > 0.36$	4380	1.0E-6
4	Earthquake 0.3g to 0.375g	120	1.0E-4
5	Earthquake 0.375g to 0.825g	4380	1.9E-5
6	Earthquake > 0.825g	4380	7.1E-7
7	Vessel Fracture	4380	1.0E-13
8	Local Core Coolant Blockage	4380	1.8E-6
9	Reactor Vessel Leak	4380	1.0E-6
10	Loss of One Primary Pump	600	1.6E-1
11	Loss of Substantial Primary Coolant Flow	8	5.0E-2
12	Loss of Operating Power Heat Removal	86	8.0E-2
13	Loss of Shutdown Heat Removal via BOP	24	8.2E-3
14	Loss of Shutdown Heat Removal via IHTS	600	1.0E-2
15	IHTS Pump Failure	600	5.0E-2
16	Station Blackout	1200	3.0E-5
17	Large Na-H2O Reaction	4380	6.0E-8
18	Spurious Scram and Transients Inadequately Handled by PCS	600	0.6
19	Normal Shutdown	600	0.6
20	Forced Shutdown	240	5.5
21	RVACS Blockage	86	1.0E-8
	TOTAL		6.398

(1) f = initiating event frequency on a per year basis ; values are given in exponential form where XE-Y = $X10^{-Y}$.
(2) t_m = Shutdown heat removal mission time in hours = expected (or mean) time required to restore to normal power operation.

Table 3. Definitions of Accident Types

Accident Type	Definition
S3	LOSHR with reactor shut down and no initial core damage
S5	LOSHR with reactor shut down but with initial partial core damage or blockage (Fermi I type accident), or with added heat due to initial transient.
P1	TOP with reactivity addition of $0.07 to $0.18
P2	TOP with either (1) reactivity addition of $0.18 to $0.36 or (2) smaller reactivity addition with loss of inherent reactivity feed back.
P3	TOP with either (1) reactivity addition >$0.36 or (2) reactivity addition of $0.18 to $0.36 with loss of inherent reactivity feedback.
P4	TOP with both reactivity addition >$0.36 and loss of inherent reactivity feedback.
P1S,….,P4S	Same as P1,….,P4, respectively, except that the accident is also accompanied by LOSHR.
F1	LOF due to pump trip with failure to scram but with successful flow coast down and inherent reactivity feedback.
F3	Same as F1, except with failure of flow coast down or failure of inherent reactivity feedback, or both.
F3S	Same as F3, except that the accident is also accompanied by LOSHR.
H2	ULOHS resulting from loss of heat removal capability with failure to scram either (1) at nominal power with loss of inherent feedback due to stuck CR's, or (2) at an elevated power of up to 125% nominal.
H3	ULOHS due to loss of heat removal capability with failure to scram either (1) at up to 125% with loss of inherent feedback or (2) at power >125%.
H1S	ULOHS with failure to scram at nominal power with successful inherent reactivity feedback but with LOSHR.
H2S, H3S	Same as H2 and H3, except that the accidents are also accompanied by LOSHR.
G3	A combined P2/F3 or P3/F1.
G4	A combined P4/F1 or P3/F3.
G1S	A combined P2/F1 or P1/F1 with LOSHR.
G3S, G4S	Same as G3 and G4, except that the accidents are also accompanied by LSHR.

Table 4. Containment Release Categories

Categories	Description
R2A	25% early core melt transient, early debris not coolable
R3	100% early core melt transient with energetic expulsion, Debris coolable, no melt-through
R4A	100% early core melt transient with energetic expulsion, early debris not coolable
R6A	No early transient, loss of SHRS and core uncovery, no late energetic expulsion
R6U	Early transient, minor core damage, otherwise same as R6A
R6S	Early transient, 25% core melt, otherwise same as R6A
R8A	No early transient, loss of SHRS and core uncovery, late energetic expulsion
RBU	Same as R6U but with late energetic expulsion
R8S	Same as R6S but with late energetic expulsion

Table 5. Public Risk from the Operation of A Prism Module

Risk Measure	NRC Goal (less than)	PRISM Performance	
		With Evacuation	Without Evacuation
Societal Risk (probability of latent cancer fatality per one year of operation, 0-10 mi)	1.9×10^{-6}	9.0×10^{-12}	1.2×10^{-11}
Individual Risk (probability of prompt fatality per one year of operation, 0-1 mi)	5×10^{-7}	$<10^{-13}$	2.7×10^{-10}

(CONPAS Ver 1.0) Diagram: SYSET.KET /User: /Date: 12 Feb 99

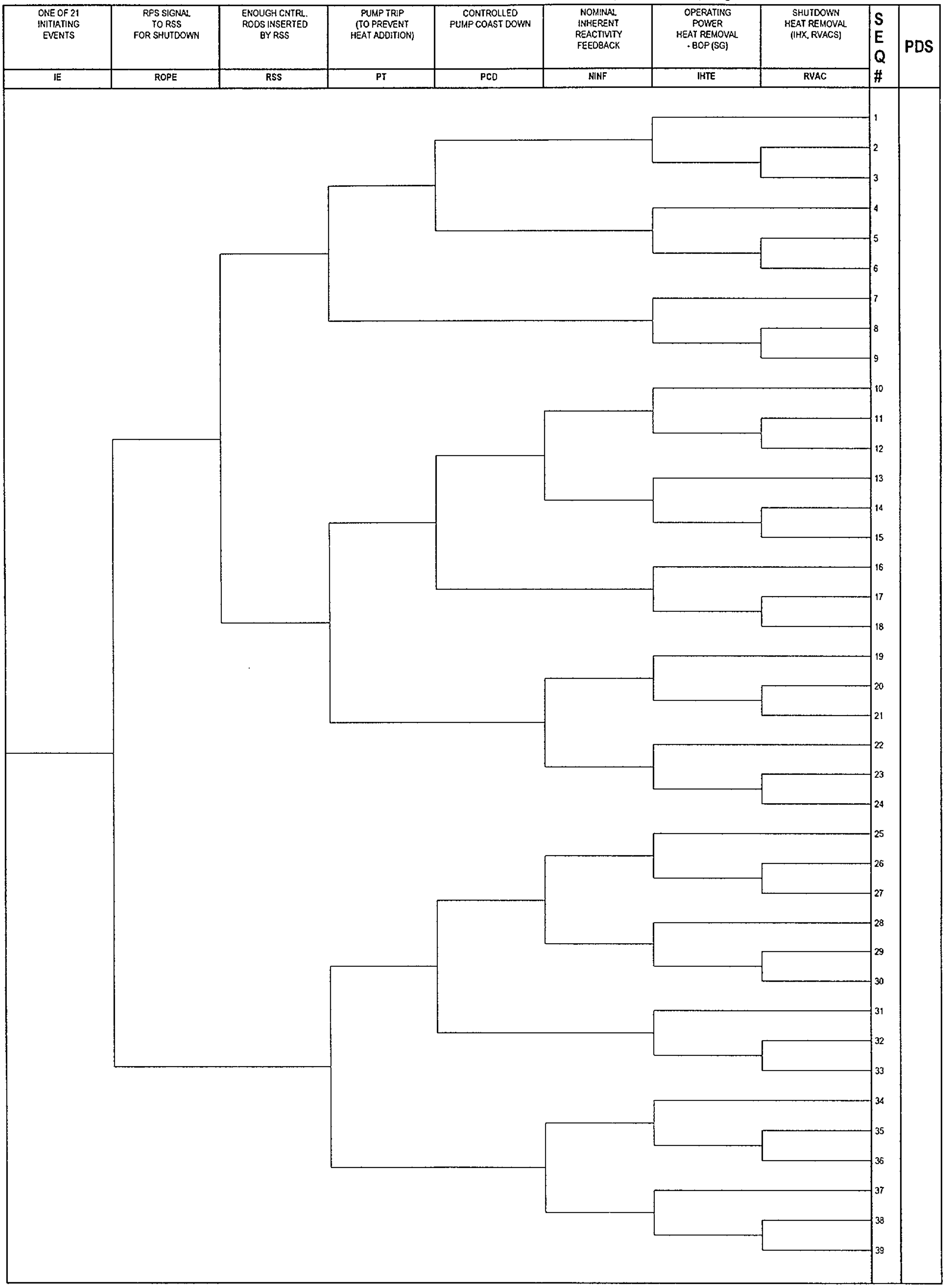

Figure 2. An Example of System Event Trees

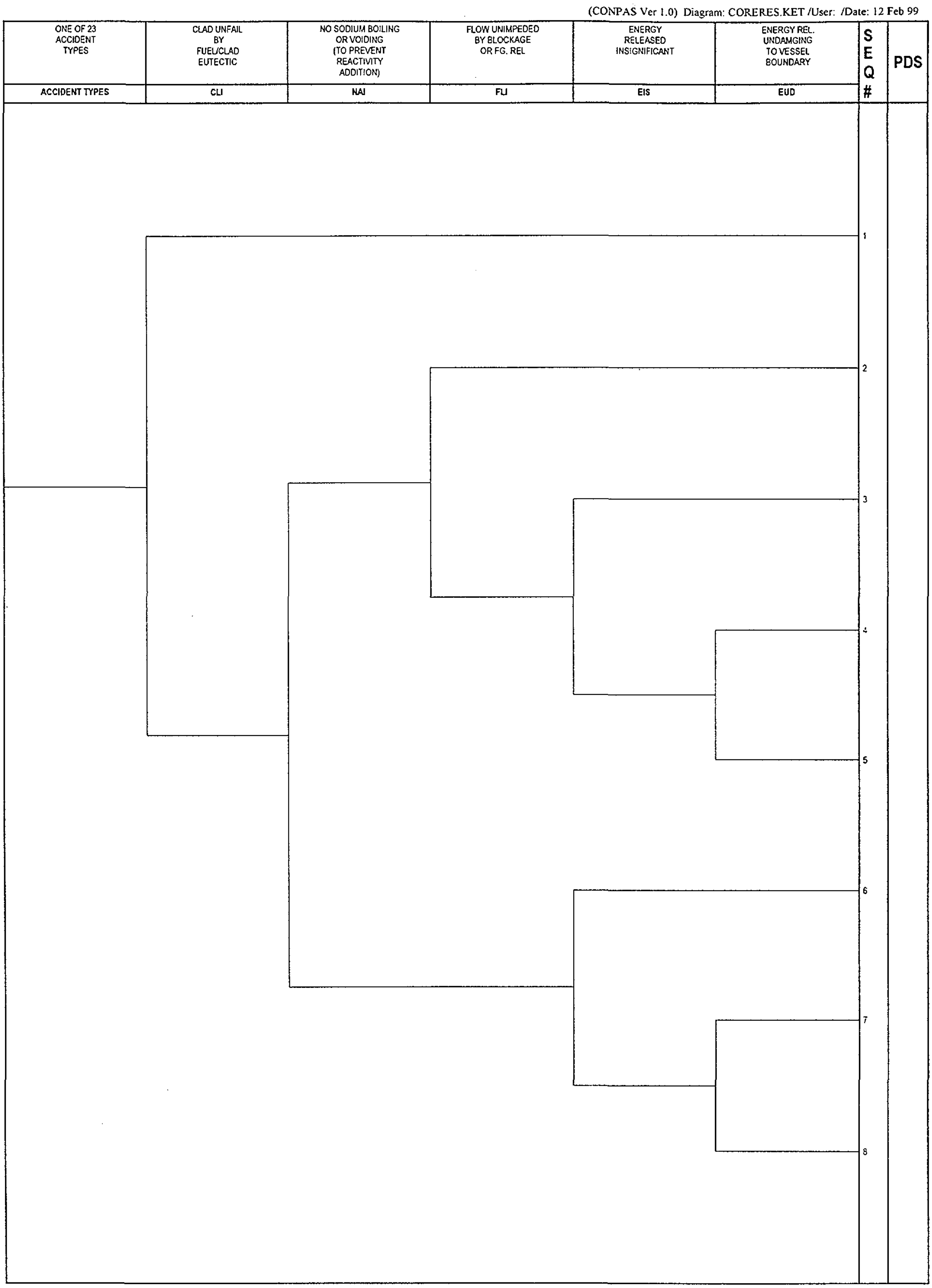

Figure 3. An Example of Core Response Event Trees

(CONPAS Ver 1.0) Diagram: CTMTRES.KET /User: /Date: 12 Feb 99

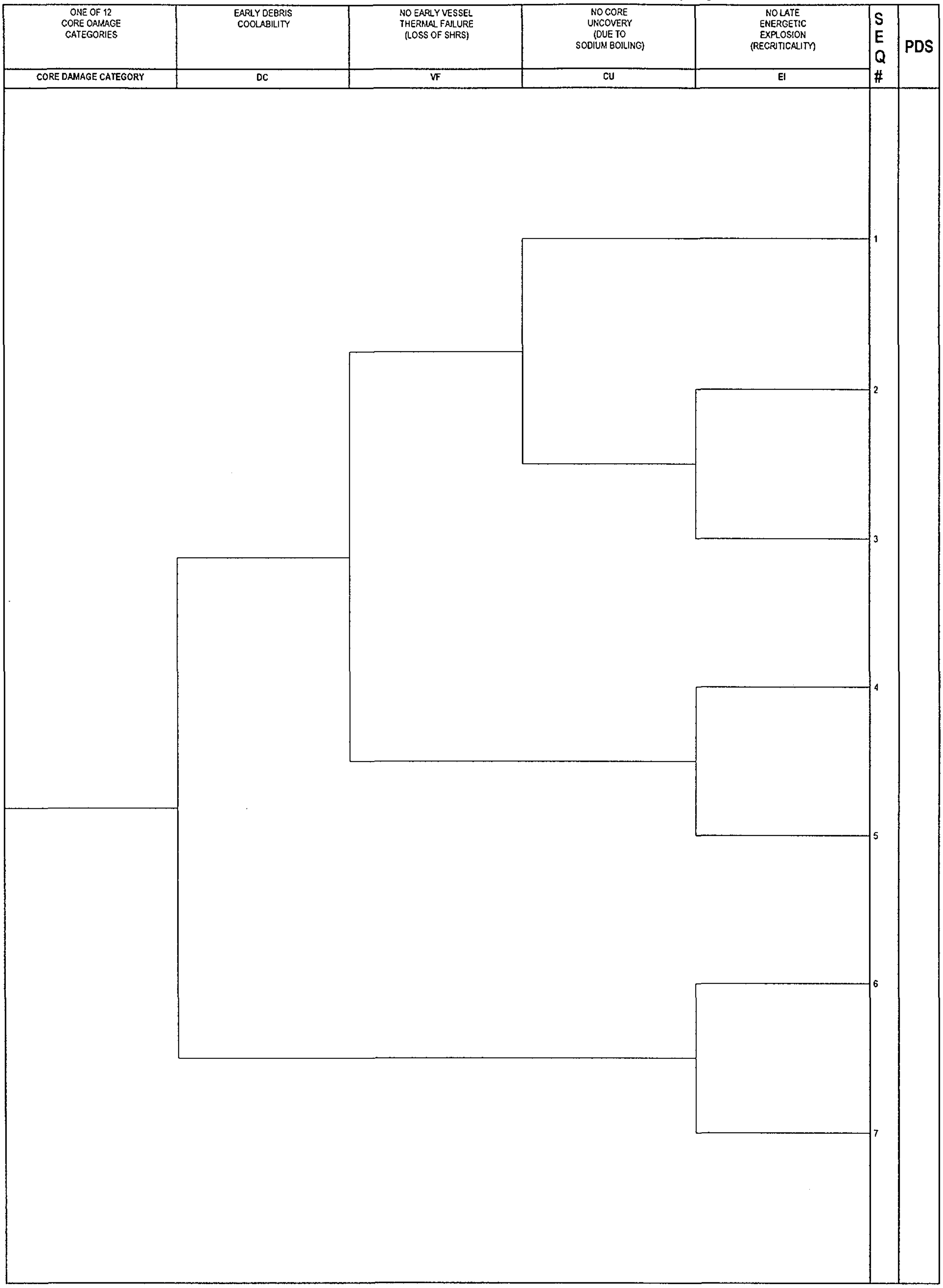

Figure 4. An Example of Containment Response Event Trees

4. RECOMMENDATIONS FOR KALIMER PRA

Since the design concept of KALIMER is similar to that of the PRISM plant developed by GE, it would be appropriate to review the PRA methodology of PRISM as the first step of KALIMER PRA. Hence, in this report, we review the PRA methodology of the PRISM plant. The PRA methodology of the PRISM plant has the same goal as that of LWR/PHWR PRA methodologies. However, owing to the differences in design, the detailed procedures and data are different from those of LWR/PHWR PRA. In addition, as the PRISM plant is in the design stage, the data are not enough for detailed PRA. So, some of the PRA methodology is not clearly defined. These aspects will be discussed in this section.

The PRA methodology of the PRISM plant consists of the following five major tasks: (1) development of an initiating event list, (2) development of System Event Trees, (3) development of Core Response Event Trees, (4) development of Containment Response Event Trees, and (5) Consequences and risk estimation. In the report, each task of PRISM PRA is reviewed and compared to the corresponding part of LWR/PHWR PSA performed in Korea. The parts that are not modeled appropriately in PRISM PRA are identified, and the recommendations for KALIMER PRA are also summarized.

4.1 RECOMMENDATIONS FROM PRISM PRA

This section summarizes the recommendations from PRISM PRA report for the further works. These recommendations are also to be considered in KALIMER PRA.

Although conservative assumptions have been used in assigning the probability of structural failures and paths leading to severe accidents in PRISM PRA, further analysis is required in the following areas to develop an information base for a more realistic assessment:

1. The risk model should be expanded to include a detailed systematic analysis of man/machine interactions following the occurrence of an initiating event. In particular, assurance that potential accident paths have been conservatively accounted for will be enhanced with explicit modeling of the effect of accident sequences on the operator's cognitive behavior and on the capability of post-accident monitoring and recovery.

2. Detailed failure modes and effects analysis (FMEA) of the reactor core, in-vessel structures, reactor and guard vessels, and other structures is recommended to uncover potential paths for loss of the inherent reactivity feedback features, loss of the heat removal functions, and dependent failures.

3. Fragility analysis is required to assess the probability of the critical failure modes identified in the above FMEA. Seismic analysis to assess the probability of failure propagation and combinations which may lead to loss of the shutdown heat removal

function is also required.

4. Man-structure interface during manufacturing, repair, inspection, and operation, and the quality assurance program for these operations should be analyzed to assess the possibility of structural defects which may propagate to serious failures due to applied stresses or man-structure interaction.

5. Detailed common cause failure analysis is needed to replace the conservative beta factor approach used in the PRA and to identify types of dependencies which may be removed by design or operating procedures.

4.2 COMPARISONS WITH LWR/PHWR PRA METHODOLOGY

Each task of PRISM PRA is reviewed and compared to the corresponding part of LWR/PHWR PSA performed in Korea. We compared the tasks based on the assumption that we will use the PRISM PRA procedures for KALIMER PRA.

It is identified that following works are required for the KALIMER PRA. Some of them are already mentioned in the previous sub-section. In this sub-section, we will explain them in detail.

- Development of Initiating Event List:

The basic approach to derive the list of initiating events is similar to that of LWR/PHWR such as the review of past PRA and the plant design. However, there are no detailed descriptions on how to derive the list of initiating events in PRISM PRA. For KALIMER PRA, the use of the master logic diagram will be a good way to derive the list of initiating events systematically based on the plant design as in LWR/PHWR PRA since there is no similar PRA except PRISM PRA.

As the initiating events depend upon the design of a NPP, the derived list of initiating events of PRISM PRA shows some unique initiating events compared to those of LWR/PHWR PSA such as "the RVACS Blockage." This list also has some special features like following:

1. the reactivity induced events and the seismic events are classified in detail, and
2. the BOP induced events are lumped as a big category like "Loss of Shutdown Heat Removal via BOP."

Such facts are due to the characteristics of PRISM plant design. That is, the design of the PRISM plant provides the better capability of withstanding seismic events compared to LWR/PHWR design. The PRISM plant also has the capability to remove the decay heat via passive ways like RVACS. So the failures of heat removal systems are not big concerns in PRISM PRA as in LWR/PHWR PRA. Since the basic design concept of KALIMER is similar to that of PRISM plant, it is expected that the similar features will appear in the initiating events of KALIMER PRA.

For the external event analysis, only the seismic events are analyzed. The other external events such as flooding and fire are to be included in the list of initiating events for the completeness.

- Development of System Event Tree:

The concept of the PRISM system event trees is similar to that of system event trees of LWR/PHWR PSA. However, there is a difference in defining the ending states of event trees. In LWR PSA, the ending state is divided into only two categories: Core Damage State and Safe State. However, in PRISM PRA, the ending states of system event trees represent not the core damage states but the accident types. It is similar to the system event tree of PHWR PSA in which the ending states of event trees are divided into 10 PDS according to the severity of accidents. In PRISM PRA, 23 accident types are defined for the ending states of system event trees. These accident types are to be re-defined for KALIMER PRA based on the design differences between KALIMER and PRISM plant.

In addition, there are several items that are not modeled appropriately in the development of system event tree for PRISM PRA such as followings. These items are to be modeled correctly in KALIMER PRA.

1. Fault Trees of Systems

   The system fault trees are built and used to derive the unavailability of systems in this stage of PRISM PRA. The fault trees of systems are built very briefly in PRISM PRA. It seems due to the lack of detailed data. If we can get more detailed data of systems, more detailed fault trees are to be developed for KALIMER PRA.

   In PRISM PRA, the reliability of passive systems such as RVACS is estimated by using the reliability block diagram. The unavailability of passive systems is estimated extremely low. At present, only the structural failure of passive systems is considered. For KALIMER PRA, we need a more complete method to assess the reliabilities of passive systems.

2. CCF Analysis

   In PRISM PRA, the probability of independent failures under nominal design conditions has been estimated to be extremely low. This raises the relative importance of dependent failures and of operating conditions outside the design envelope as risk contributors. However, as described in Section 4.1, the CCF of systems is treated very briefly in PRISM PRA by using the beta factor method.

   Since the CCF is the major contributor to the unavailability of systems, especially for the systems with multiple trains, we have to model the CCF in detail by using the methods such as MGL and/or alpha factor method for KALIMER PRA.

3. Human Reliability Analysis (HRA)

The exact role and the types of human actions are not described in PRISM PRA. Only the recovery actions by operator are suggested as the future work. Since the human error is also a major contributor to the risk of the plant, we need to assess human error in detail based on the HRA methodology of LWR/PHWR PRA.

4. Digital Instrument and Control (I&C) Reliability Analysis

The reactor protection system of KALIMER consists of digital I&C systems. So, we have to assess the reliabilities of those systems. However, the methodology to assess the reliability analysis of digital I&C systems is not established yet. This problem is to be solved to assess the risk of KALIMER appropriately.

- Development of Core Response Event Tree:

By using Core Response Event Tree, 23 accident types of system event trees are divided into 12 core damage categories according to the transient phase of the core. In general, there are no event trees like the core response event trees in LWR/PHWR PSA which identify the possible core scenarios until neutronic shutdown is accomplished.

If we use the same concept as the Core Response Event Tree in KALIMER PRA, we have to consider a following point in developing the Core Response Event Tree for KALIMER PRA:

The molten core behavior under accident conditions might be different between the PRISM and KALIMER plant owing to the differences in fuel materials. The reference PRISM core uses a I-Pu-Zr metal fuel. On the other hand, KALIMER core uses Na-U235-U238 metal fuel. Such differences in fuel material can result in a different melting temperature and different fission products. Again, it can cause different behavior of the core after the accidents. These aspects are to be analyzed for KALIMER PRA.

- Development of Containment Response Event Tree:

The concept of a containment response event tree is the same as that of LWR/PHWR PRA. Containment response event tree models the physical phenomena such as the behavior of debris, vessel failure, sodium voiding, etc. As the result of analysis, we will get a quantitative description of the nature of release.

However, owing to the design differences in both reactor types, the containment response event tree is built a little bit differently. In LWR/PHWR PRA, before the construction of Containment Event Tree (CET), we develop PDS event trees to model the operation of safety systems such as containment spray and hydrogen igniter, etc. After that we develop the CET which models physical phenomena. However, there are no PDS event trees in PRISM PRA due to the design differences in both reactor types.

Since KALIMER has the similar containment design with PRISM, it is expected that the containment response event tree of PRISM PRA can be used for KALIMER PRA.

- Consequences and Risk Estimation:

There are no big differences in evaluating the consequences of accidents since this procedure is independent from the design of reactors. Only the differences will be the site characteristics such as the site population distribution and meteorological data, etc.

The occurrence of earthquake events depends upon the geological characteristics of the site. Since the main contributor to CDF in PRISM PRA is identified as a large earthquake (>0.825 g ground acceleration), the geological characteristics of the site are to be incorporated into KALIMER PRA to assess seismic events appropriately.

As explained above, some parts of PRSIM PRA methodology are different from those of conventional PRA methodologies. Table 6 summarizes the PRA procedures for PRISM PRA, LWR/PHWR PRA. The proposed procedures for KALIMER PRA are also shown in Table 6. Table 7 summarizes the characteristics of each procedures for PRISM PRA, LWR/PHWR PRA.

4.3 COMPUTER CODES AND DATA

In addition to above, such as the computer codes and data used in the PRISM PRA are also compared to the computer codes and data available in Korea. The compared results are summarized in Table 8-9.

The PSA codes, KIRAP [8] and CONPAS [9], developed by KAERI can be used for PRA works such as fault and event tree analysis. KAERI also have accident analysis codes such as SSC-K [11] and CONTAIN-LMR [12] which can be used instead of SASSYS, ARIES and CONTAIN used in PRISM PRA.

Since PRISM PRA was performed in '80, the reliability data is old one such as NPRDS of '80. So, it is expected that the use of the updated reliability data such as ALWR URD [10] is desirable for KALIMER PRA. Site specific data such as the fragility curve is to be derived for the specific site for KALIMER PRA. Some reliability data are not available at present, e.g. the reliability data for digital I&C systems. This problem is to be solved as well.

In addition, many expert judgements are used in PRISM PRA due to the lack of detailed data. So, we need to get more detailed information about those areas.

There is an additional difference between KALIMER and PRISM PRA. That is due to different regulatory environments. Up to now, the safety goal and the role of PRA are not clearly defined for LMFBR in Korea as for the PRISM plant in U.S.A. Only the CDF and large radioactive release frequency are assumed to be lower than 1.0E-6/year and 1.0E-7/year, respectively by the designer [13]. The role and scope of PRA will be changed according to such regulatory policies. Hence, the regulatory policy on KALIMER is to be determined before the PRA of KALIMER.

The PRA methodology of PRISM plant is briefly reviewed and compared to the corresponding part of LWR/PHWR PSA performed in Korea. The parts that are not modeled appropriately are identified, and the recommendations for KALIMER PRA are stated above. Since we do not have any experience in LMFBR PRA, it will be good way to start KALIMER PRA based on PRISM PRA methodology. In technical aspects, most parts of the PRA methodology of PRISM can be applied to the KALIMER PRA with appropriate considerations regarding to above items.

Table 6. **Comparison of Procedures for PRISM, LWR/PHWR and KALIMER PRA**

Level of PRA	Risk Model Element	PRISM PRA	LWR/PHWR PRA	KALIMER PRA
Level 1	Initiating Events	- FTA - Availability analysis	- MLD - Review of past PRA	- MLD - Review of past PRA
	System Event Trees	- FTA/ETA - Reliability block diagrams - Mathematical availability models - Fragility analysis - Dependency analysis	- FTA/ETA - CCF analysis - HRA - Fragility analysis	- FTA/ETA - CCF analysis - HRA - Fragility analysis
Level 1 / Level 2	Core Response Event Trees	- ETA - Parametric evaluations of accident energetics - Bounding analysis of reactivity worth and rate	N/A	- ETA - Parametric evaluations of accident energetics - Bounding analysis of reactivity worth and rate
Level 2	Containment Response Event Trees	- Thermal analysis - Bounding calculations	- CET - Thermal analysis	- CET - Thermal analysis - Bounding calculations
Level 3	Consequences	- Core inventory and release fraction - Individual and societal risk analysis		

- FTA: Fault tree analysis
- MLD: Master Logic Diagram
- ETA: Event Tree Analysis
- CCF: Common Cause Failure
- HRA: Human Reliability Analysis
- CET: Containment Event Tree

Table 7. Characteristics of PRISM and LWR/PHWR PRA Methodologies

Risk Model Element	PRISM PRA	LWR & PHWR PRA	Comments
Initiating Events	- 21 Initiating Events - Reactivity events - Earthquake events - Loss of heat removal	- LOCA group - Transient group - ATWS	
System Event Trees	- 23 accident types	- 2 core damage states in LWR PSA and 10 plant damage status in PHWR PSA	
Core Response Event Trees	- 12 core damage categories	N/A	
Containment Response Event Trees	- 9 containment release categories - Physical phenomena	- Physical phenomena and operation of safety systems	There is no PDS ET in PRISM PRA.
Consequences & Risk	Latent and prompt fatality		

Table 8. Comparison of Computer Codes for PRISM, LWR/PHWR and KALIMER PRA

Risk Model Element	PRISM PRA	LWR/PHWR PRA	KALIMER PRA
Initiating Events	N/A	- KIRAP [8]	- KIRAP
System Event Trees	- FRANCALC-1	- KIRAP - CONPAS [9]	- KIRAP - CONPAS
Core Response Event Trees	- SASSYS - ARIES	N/A	- SSC-K [11]
Containment Response Event Trees	- CONTAIN	- MAPP	- CONTAIN-LMR [12]
Consequences	- MACCS - ORIGEN		

Table 9. Comparison of Data Sources for PRISM, LWR/PHWR and KALIMER PRA

Risk Model Element	PRISM PRA	LWR/PHWR PRA	KALIMER PRA
Initiating Events	- NPRDS - CRPRA - GESSAR site seismic frequency	- ALWR URD/DARA - Site seismic frequency	- ALWR URD - PRISM PRA - Site seismic frequency
System Event Trees	- Reactor Safety Study (WASH-1400) - NPRDS - CRPRA - LLNL generic fragility data	- ALWR URD/DARA - Site specific fragility data	- PRISM PRA - ALWR URD - Site specific fragility data
Core Response Event Trees	- WASH-1400 - Metal fuel data handbook - Appendix E of this PSID: Analysis of BDBE	N/A	- PRISM PRA - Metal fuel data handbook - Analysis of BDBE for KALIMER
Containment Response Event Trees	- Nuclear Systems Material Handbook	- MATPRO [14]	- Nuclear Systems Material Handbook
Consequences	- NUREG-1150 - GESSAR site population distribution and meteorological data	- NUREG-1150 - Site population distribution and meteorological data	- NUREG-1150 - Site population distribution and meteorological data

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<table>
<tr><td colspan="6">BIBLIOGRAPHIC INFORMATION SHEET</td></tr>
<tr><td colspan="2">Performing Org. Report No.</td><td colspan="2">Sponsoring Org. Report No.</td><td>Standard Report No.</td><td>INIS Subject Code</td></tr>
<tr><td colspan="2">KAERI/AR-524/99</td><td colspan="2"></td><td></td><td></td></tr>
<tr><td colspan="2">Title / Subtitle</td><td colspan="4">Review of PRA Methodology for LMFBR</td></tr>
<tr><td colspan="2">Project Manager and Dept.</td><td colspan="4">Yang, Joon-Eon (Integrated Safety Assessment Team)</td></tr>
<tr><td colspan="2">Researcher and Dept.</td><td colspan="4"></td></tr>
<tr><td>Publication Place</td><td>Taejon</td><td>Publisher</td><td>KAERI</td><td>Publication Date</td><td>1999.2.</td></tr>
<tr><td>Page</td><td>42p</td><td>Ill. & Tab.</td><td>Yes(√), No( )</td><td>Size</td><td>30 cm</td></tr>
<tr><td>Note</td><td colspan="5"></td></tr>
<tr><td>Classified</td><td colspan="2">Open(√), Restricted( ),
__Class Document</td><td>Report Type</td><td colspan="2">State of the Art Report</td></tr>
<tr><td colspan="2">Sponsoring Org.</td><td></td><td>Contract No.</td><td colspan="2"></td></tr>
<tr><td colspan="6">Abstract (15-20 Lines)</td></tr>
<tr><td colspan="6">Probabilistic Risk Assessment (PRA) has been widely used as a tool to evaluate the safety of NPPs (Nuclear Power Plants), which are in the design stage as well as in operation. Recently, PRA becomes one of the licensing requirements for many existing and new NPPs. KALIMER is a Liquid Metal Fast Breeder Reactor (LMFBR) being developed by KAERI. Since the design concept of KAIMER is similar to that of the PRISM plant developed by GE, it would be appropriate to review the PRA methodology of PRISM as the first step of KALIMER PRA. Hence, this report summarizes the PRA methodology of PRISM plant, and the required works for the PSA of KALIMER based on the reviewed results. The PRA technology of PRISM plant consists of following five major tasks: (1) development of initiating event list, (2) development of System Event Tree, (3) development of Core Response Event Tree, (4) development of Containment Response Event Tree, and (5) Consequences and risk estimation. The estimated individual and societal risk measures show that the risk from a PRISM module is substantially less than the NRC goal. Each task is compared to the PRA methodology of Light Water Reactor (LWR)/Pressurized Heavy Water Reactor (PHWR). In the report, each task of PRISM PRA methodology is reviewed and compared to the corresponding part of LWR/PHWR PSA performed in Korea. The parts that are not modeled appropriately in PRISM PRA are identified, and the recommendations for KALIMER PRA are stated.</td></tr>
<tr><td colspan="2">Subject Keywords (about 10 words)</td><td colspan="4">LMFBR, Probabilistic Safety Assessment, Safety Goal, Design Improvement</td></tr>
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서 지 정 보 양 식

수행기관보고서번호	위탁기관보고서번호	표준보고서번호	INIS 주제코드
KAERI/AR-524/99			

제목 / 부제	액체 금속로의 확률론적 위해도 평가 방법 검토
연구책임자 및 부서명 (AR, TR 등의 경우 주저자)	양준언 (종합안전평가팀)
연 구 자 및 부 서 명	

출 판 지	대전	발행기관	한국원자력연구소	발행년	1999.2.
페 이 지	42 p.	도 표	있음(√),없음( )	크 기	30 cm
참고사항	원자력 연구개발 중장기 과제				
비밀여부	공개(√),대외비( ),__급비밀		보고서종류	기술현황보고서	
연구위탁기관			계약 번호		

초록 (15-20 줄)

확률론적 위해도 평가방법 (PRA, Probabilistic Risk Assessment)은 설계 또는 운전 중인 원자력 발전소의 안전성을 종합적으로 파악하기 위한 방법으로 근래에 들어 PRA 의 수행은 많은 발전소의 인허가 요건이 되고 있다. 한국 원자력연구소에서는 액체금속로인 KALIMER 를 개발하고 있다. 세계적인 인허가 요건에 따라 추후에 KALIMER 의 PRA 도 수행하여야 할 것으로 예상된다. 기존의 PRA 방법은 경수로나 중수로를 중심으로 정립이 되어 있으나, 액체금속로의 경우에 있어서는 그 설계가 전혀 상이하며 사고 진행도 다르므로 액체금속로를 위한 새로운 PRA 방법이 필요하다. KALIMER 의 기본 설계가 GE 에서 개발한 PRISM 발전소와 유사하므로 본 보고서에서는 PRISM 발전소의 PRA 방법론을 정리하고 이 방법론의 KALIMER PRA 에 대한 적용 가능성에 대하여 살펴본다. PRISM PRA 방법은 기본적으로 다음의 5 가지 과정으로 이루어져 있다: (1) 초기 사건의 도출, (2) 계통 사건 수목의 구축, (3) 노심 대응 사건 수목의 구축, (4) 격납용기 대응 사건 수목의 구축 및 (5) 방출 영향과 위해도 평가. PRISM PRA 의 결과에 따르면 PRISM 발전소의 위해도는 기존 경/중수로에 비하여 매우 적은 것으로 나타나고 있다. 본 보고서에서는 PRISM PRA 의 각 과정에 대해 경/중수로 PRA 방법론과 비교를 하여 그 차이점을 살펴보았으며, 또한 KALIMER PRA 를 수행하기 위하여 필요한 작업들을 정리하였다. 아울러, KALIMER PRA 를 위하여 필요한 컴퓨터 코드 및 자료도 정리하였다.

주제명키워드 (10 단어내외)	액체금속로, 확률론적 안전성 평가, 안전 목표, 설계 개선