PRISM INTERNAL EVENTS PRA MODEL DEVELOPMENT AND RESULTS SUMMARY

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GE Hitachi Nuclear Energy (GEH) has teamed with Argonne National Laboratory to perform research and development of next-generation Probabilistic Risk Assessment (PRA) methodologies and analyses for the modernization of an advanced non-light water reactor (LWR) PRA. This effort builds upon a PRA developed in the early 1990s for GEH's PRISM sodium fast reactor. This project has four main tasks: Internal Events development which models the risk from the reactor plant for hazards occurring at-power internal to the plant; an All Hazards review to analyze the risk at a high level from external hazards such as earthquakes and high winds; an All Modes review to understand the risk at a high level from operating modes other than at-power; and Risk Insights to integrate the results from each of the three phases above.

In the first phase of the project, GEH and Argonne used and adapted proven PRA methodologies and techniques to build a modern non-LWR internal events PRA. This paper presents the results for the internal events PRA. Non-LWR methodologies developed in this project used the requirements from the trial-use of the ASME/ANS non-LWR PRA standard to build an integrated risk model that incorporates the probability and consequence of various postulated events for the PRISM plant design. The probability models are developed first using basic events defined by Initiating Events, Data, and Human Reliability analysis. Probabilities of events and sequences are then quantified using logic models developed in System Analysis and Event Sequence Analysis. The consequences of postulated events are evaluated using thermal-hydraulic calculations in Success Criteria and Mechanistic Source Term analysis, as well as radionuclide dispersion calculations in Radiological Consequence Analysis. The final analysis, Risk Integration, combines the results from the probability modeling and the consequence calculations to develop a complete picture of accident risk for the PRISM design.

I. INTRODUCTION

This paper provides a summary of the internal events at-power for the Power Reactor Inherently Safe Module (PRISM) Probabilistic Risk Assessment (PRA). The overall objective of the internal events PRA is to provide a quantitative measure of risk by calculating the likelihood of potential radiological consequences. Innovative methodologies developed during the project were used to help assess the PRISM design against the NRC Quantitative Health Objectives (QHOs).

In 2013, the American Society of Mechanical Engineers (ASME) and the American Nuclear Society (ANS) published a PRA standard for advanced non-LWR designs [Ref. 1]. The standard, currently available for trial-use, provides technology-neutral requirements for a full scope PRA. The analyses summarized in this paper adhered[1] to the requirements of the standard. This PRA is one of the first to comprehensively utilize the trial-use standard.

I.A. Project Overview

GE Hitachi Nuclear Energy (GEH) has teamed with Argonne National Laboratory to perform research and development (R&D) of next-generation Probabilistic Risk Assessment (PRA) methodologies for the modernization of an advanced non-light water reactor (LWR) PRA. This project is supported by the Department of Energy under Award Number DE-NE0008325. This effort builds upon a PRA developed in the early 1990s for GEH's PRISM sodium fast reactor.

Figure 1 provides an overview of the project's deliverables and benefits. This next-generation PRA will allow a risk-informed R&D prioritization option going forward. GEH, with its extensive experience in PRA, is leading the research efforts and is leveraging Argonne's unique expertise in key areas of advanced PRA analysis.

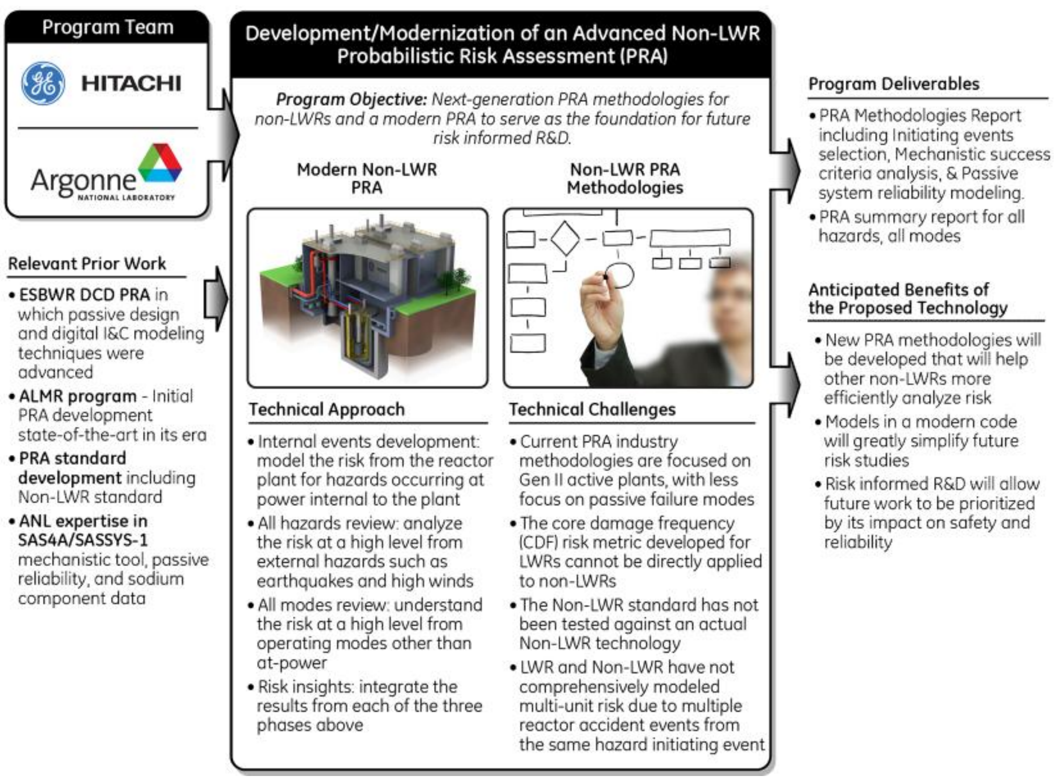

Fig. 1. PRISM PRA Modernization Project Overview

[1] In a few instances, a certain requirement could not be met due to either project scope limitations or potential weaknesses in the trial use standard. Feedback will be provided to the standard committee for these instances.

I.B. PRISM Plant Overview

The PRISM is a pool-type, metal-fueled, small modular Sodium Fast Reactor (SFR). PRISM employs passive safety, digital instrumentation and control, and modular fabrication techniques to expedite plant construction. The PRISM has a rated thermal power of 840 MWth and an electrical output of 311 MW. Each PRISM module has an intermediate sodium loop that exchanges heat between the primary sodium coolant from the core with water/steam in a sodium-water steam generator. The steam from the sodium-water steam generator feeds a conventional steam turbine. A diagram of the PRISM nuclear steam supply system (NSSS) is shown in Figure 2.

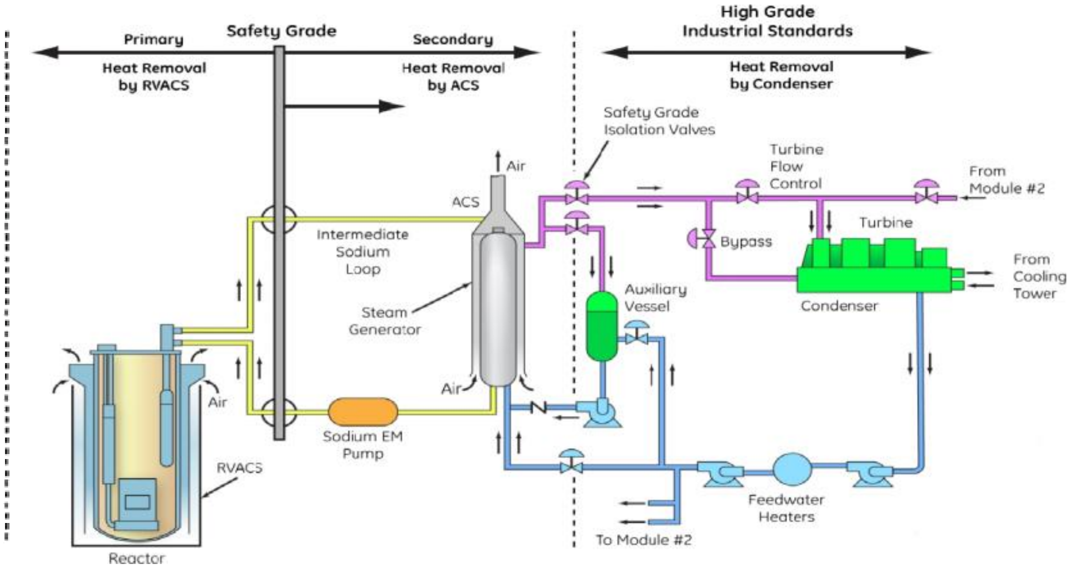

Fig. 2. PRISM nuclear steam supply system

Two reactor units are paired to form one power block. A PRISM power block is shown in Figure 3. The power block supplies steam for one 622-MW turbine-generator. The commercial PRISM plant achieves a high capacity factor by utilizing six reactor modules and their associated steam generating systems arranged in three identical power blocks. An unplanned outage in one PRISM module or power block does not impact the plant electrical output as dramatically as it does in a large single-unit site.

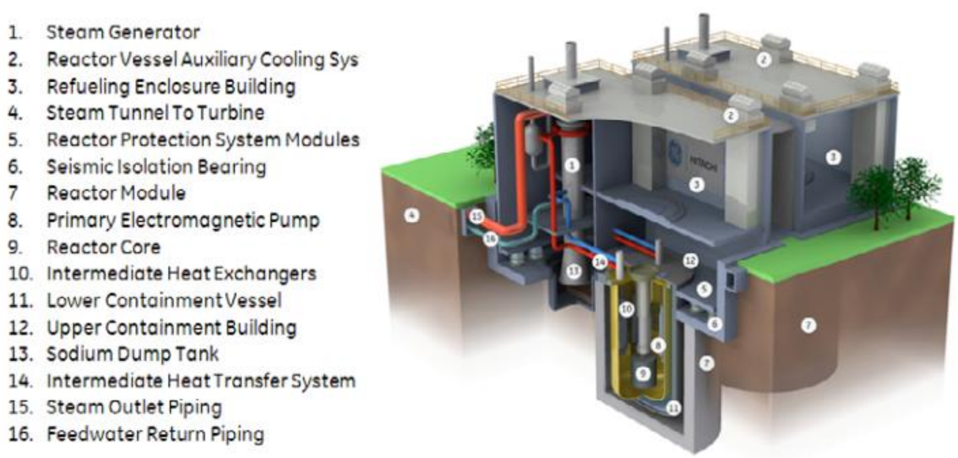

Fig. 3. PRISM Power Block

Plant electrical output can be tailored to utility needs by the modular addition of power blocks. This modularity allows expansion from one power block to as many as desired by the utility on one site.

II. METHODOLOGY OVERVIEW

The PRISM PRA was performed in an iterative manner, yet there was a linear flow of information from one element to the next. The relationships between PRA tasks are summarized in Ref. 1 and there is a general delineation between tasks associated with modeling the frequencies of accident conditions and those evaluating the consequences of the accidents. A visual representation of these relationships is provided in Figure 4:

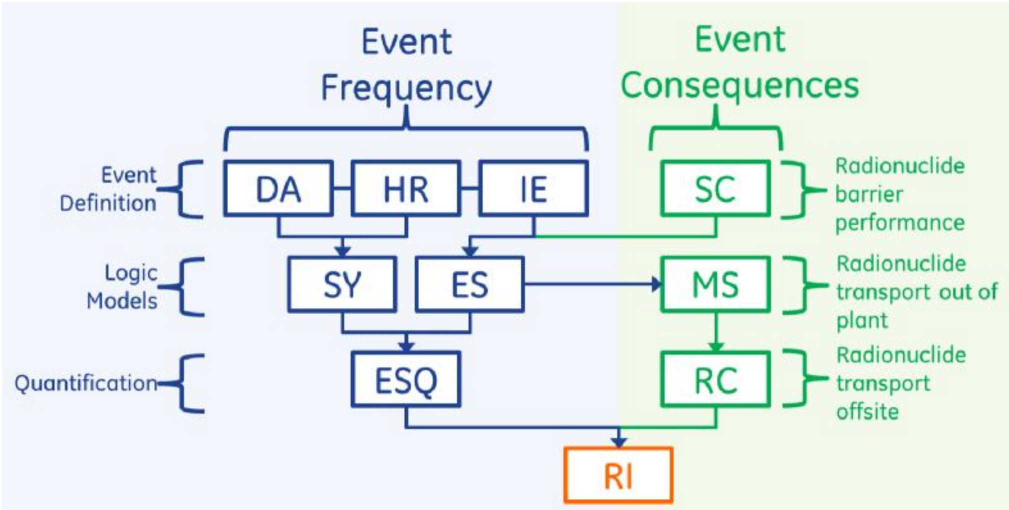

Fig. 4. PRISM PRA Methodology

Many of these elements were developed using methodologies typical to LWR PRA, however several new or modified approaches were utilized especially in the Success Criteria (SC) and Mechanistic Source Term (MS) elements. For Initiating Events (IE), identification relied upon LWR experience, Non-LWR experience and an inductive systems-based approach. In the Data Analysis (DA) task, the data pedigree was established using a prioritized data source selection process in which design specific data was compared against more plentiful generic nuclear and non-nuclear sources. Finally, concerning the HRA task, although Non-LWRs are less reliant on operator actions, pre and post initiator HRA is still an important PRA element and can identify potential design weaknesses. Human failure events were identified and error probabilities applied accordingly.

For the Event Sequence (ES) analysis, three general groups of event trees were developed to analyze event progression from IE to release:

- Protected: sequences in which active reactivity insertion is successful via the control rods
- Unprotected: sequences where control rod insertion fails and inherent reactivity feedbacks or the ultimate shutdown system must succeed to satisfy the reactivity control safety function
- Confinement: analyze the various radionuclide barrier success and failure combinations for both protected and unprotected sequences

The identified active and passive mitigating system functions and features from the ES element were then modeled in the Systems Analysis (SY) task. For active systems, traditional fault tree analysis was performed, while passive systems and features were modeled using state-of-the-art passive system reliability modeling techniques.

The overall Success Criteria (SC) for the PRISM design defined the combinations of barrier and mitigating systems successes needed to prevent release, based on the identification of possible release categories. Each release category consists of a quantified level of fuel barrier damage and a combination of intact and failed confinement release barriers. For each event sequence modeled, plant parameters are defined with various thresholds that represent the different release categories that are possible for that sequence. Based on this analysis, the PRA safety functions were defined for PRISM as shown in Figure 5.

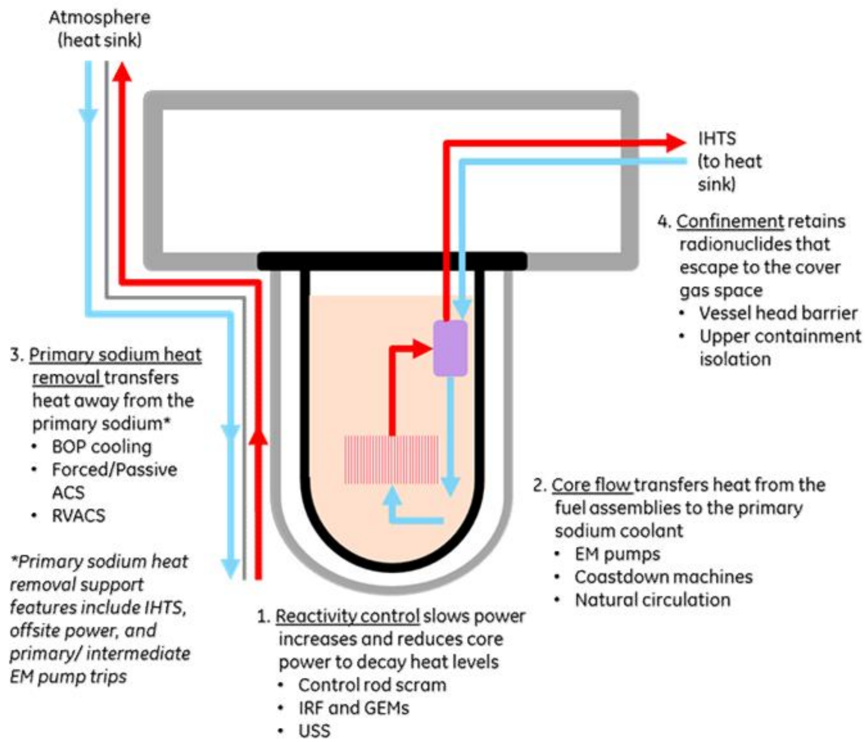

Fig. 5. PRISM PRA Safety Functions[2]

The Mechanistic Source Term (MS) and Radiological Consequence (RC) tasks analyzed radionuclide progression for the various event sequences from the fuel to offsite through the following radionuclide barriers:

1. Fuel: The metal fuel retains many radionuclides (i.e. plutonium, neptunium) within its matrix as long as the fuel has not melted.
2. Cladding: The fuel cladding provides a barrier for gaseous fission products (i.e. xenon, krypton) as long as the cladding is intact.
3. Sodium: The sodium coolant acts as a third radionuclide barrier by retaining fission products either by plate-out, chemical solubility or adsorption mechanisms.
4. Vessel: The reactor vessel is the radionuclide barrier for fission gases that are released by the sodium to the cover gas space with cladding failure.
5. Containment: The containment is the final radionuclide barrier and retains fission products with vessel leak-by failure (e.g. head seal leak).

The final step, Risk Integration (RI), combines the results of the Event Sequence Quantification (ESQ) and the Radiological Consequence analysis. For each release category identified in the PRA, the ESQ gives an annual probability of occurrence, and the RC gives the consequence of occurrence. Combining these two elements gives quantitative measures of risk that can then be compared to health goals for site workers and the public. These results are presented in the section that follows.

[2] Definitions for acronyms used in this figure: BOP: Balance of Plant, ACS: Alternate Cooling System, RVACS: Reactor Vessel Alternate Cooling System, IHTS: Intermediate Heat Transport System, EM Pump: Electromagnetic Pump, IRF: Inherent Reactivity Feedback, GEMs: Gas Expansion Modules, USS: Ultimate Shutdown System

III. RESULTS SUMMARY

III.A. Quantitative Health Objectives

Risk metrics developed for the PRISM PRA revolve around the release categories defined in the Mechanistic Source Term element. The overall plant risk can be described using a number of metrics, where each metric is essentially a frequency-weighted average of the consequences of postulated events in a given release category. Traditional risk metrics, such as core damage frequency (CDF), are not meaningful for PRISM or other non-LWR designs (as recognized by the trial use PRA standard) since they assume light water reactor core, reactor, and containment design features. Therefore, PRISM's risk must be expressed directly in offsite consequence measures as described below.

The ASME/ANS Non-LWR PRA standard [Ref. 1] uses technology neutral risk metric terms, and provides the following discussion and background for this approach:

> *The current LWR PRA standards are based on risk metrics such as core damage frequency (CDF) and large early release frequency (LERF) that have been defined in terms of LWR-specific characteristics. CDF and LERF are also known as "surrogate risk metrics" for a more complete set of risk metrics that are produced in a Level 3 PRA. For advanced non-LWRs, which include diverse reactor types such as HTGRs, LMRs, and other advanced Generation IV reactor concepts, a technology-neutral approach has been adopted in consideration of the fact that a technology-neutral definition of core damage does not exist. The risk metrics used with this technology-neutral approach are the standard risk metrics used in LWR Level 2 and Level 3 PRAs.*

Three major offsite consequence-related goals are established based on the NRC Safety Goal Policy Statement [Ref. 3]. These QHOs are defined in the sections that follow along with a presentation of the PRISM risk results. Note that these results are for one PRISM power block--two reactor modules connected to one balance of plant--and consider single and common cause initiating events whose event sequences occur asymmetrically[3]. Results do not include external hazards or potential hazards during other plant operating states.

III.B. Individual Risk QHO and PRISM Results

QHO: The risk of prompt fatalities that might result from reactor accidents to an average individual in the "vicinity" of a nuclear power plant should not exceed one tenth of one percent (0.1%) of the sum of "prompt fatality risks" resulting from other accidents to which members of the U.S. Population are generally exposed. [Ref. 3]

As noted in the Safety Goal Policy statement, "vicinity" is defined as the area within 1.61 km (1 mile) of the plant site boundary. "Prompt Fatality Risks" are defined as those risks to which the average individual residing in the vicinity of the plant is exposed to as a result of normal daily activities. "Other accidents" are the sum of risks that result in fatalities from such activities as driving, household chores, occupational activities, etc.

For this evaluation, the sum of prompt fatality risks is taken as the U.S. accidental death risk value of 39.1 deaths per 100,000 people per year [Ref. 4]. The PRISM internal events at-power individual risk is represented in the frequency-consequence curve shown in Figure 6. This curve shows the QHO goal and the prominent PRISM internal event release category consequence results (the blue data points in the figure).

[3] Concurrent event sequences are evaluated separately in the multi-unit analysis.

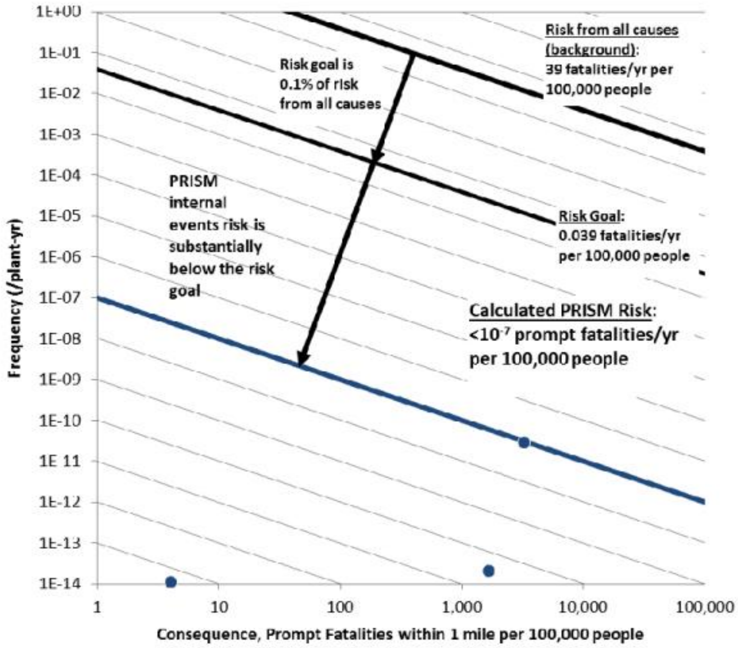

Fig. 6. Comparison of Individual Risk to Risk Goal

III.C. Societal Risk QHO and PRISM Results

QHO: The risk of cancer fatalities that might result from nuclear power plant operation to the population in the area "near" a nuclear power plant should not exceed one tenth of one percent (0.1%) of the sum of the "cancer fatality risks" resulting from all other causes to which members of the U.S. Population are generally exposed. [Ref. 3]

As noted in the Safety Goal Policy Statement, "near" is defined as within 16.1 km (10 miles) of the plant. The "cancer fatality risk" is taken as 169 deaths per 100,000 people per year based upon 1986 statistics [Ref. 5]. The PRISM internal events at-power societal risk is represented in the frequency-consequence curve in Figure 7. This curve shows the QHO goal and the prominent PRISM internal event release category consequence results (the blue data points in Figure 7).

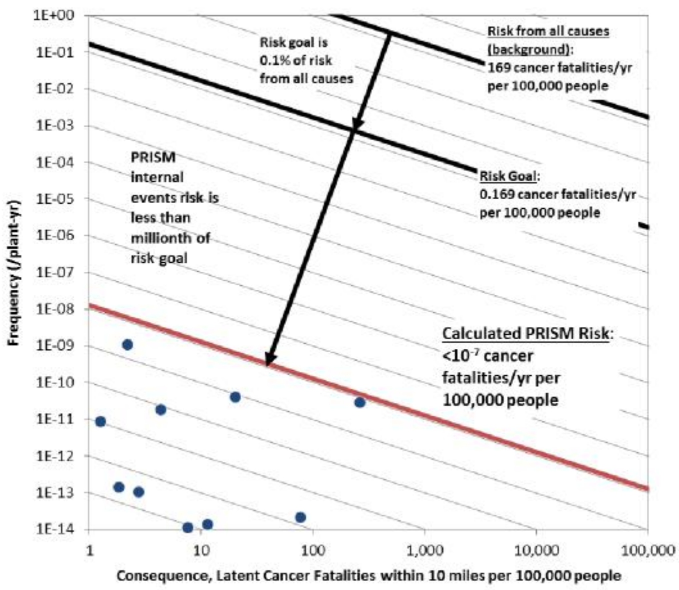

Fig. 7. Comparison of Societal Risk to Risk Goal

III.D. Multi-unit Study Results

A Multi-Unit PRA (MUPRA) was also developed for the internal events at-power phase. This was a simplified study aimed at determining the additional risk due to multi-unit events for a PRISM power block with two identical reactor units. The methodology for this study is patterned after the method outlined in IAEA draft guidance [Ref. 6] and is intended to satisfy the requirements in the ASME/ANS Non-LWR PRA standard [Ref. 1]. The MUPRA scope includes single unit and common cause initiating events which develop into concurrent event sequences.

The results from this study demonstrate that the impact from concurrent event sequences is negligible compared to the baseline results. MUPRA results in a 0.02% and 0.04% increase in individual and societal risk respectively.

III.E. Results Comparison

Although the results cannot be directly compared to the LWR metrics of CDF and LERF, a direct comparison can be made to other nuclear plants that have undergone a Level 3 PRA. Based on a Level 3 comparison, PRISM provides a large margin of risk reduction over the current generation of advanced LWRs which already possess very low risk.

IV. CONCLUSIONS

The results show that PRISM's inherent safety features provide a large reduction in both individual and societal risks over existing LWR plant designs, which are the two health objectives[4] reported by most PRAs. Such a large reduction is realized because of several PRISM design features, including:

- Passive decay heat removal via the Reactor Vessel Alternate Cooling System which has no moving mechanical parts and whose operation requires no actuation signals or external power sources
- Strong inherent negative reactivity feedback for core reactivity control
- Primary sodium coolant relatively large thermal capacity and low pressure.
- Release following core damage is minimal due to absorption of the release radionuclides into the primary sodium coolant.

Many of the more benign release events included in the PRISM PRA have not been included in the scope of previous PRAs. Moreover, the PRISM PRA conservatively analyzes and assumes a release for any amount of fuel damage using conservative dose evaluations, and does not credit any offsite evacuation in calculating public dose. If more realistic dose and leakage analysis were performed, the calculated individual and societal risks would be lower.

The PRISM internal events at-power analysis is one of the first Generation IV PRAs developed with modern PRA methodologies and standards. This model along with the full scope (all hazards, all modes) analyses will allow PRISM to be truly risk informed in its design and licensing.

[4] For LWR plant PRAs, the risk metric Large Early Release Frequency (LERF) is usually treated as a surrogate for the individual risk health objective. The Core Damage Frequency (CDF) metric is usually treated as a surrogate for the societal risk health objective.

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