Release

September 26, 2002

MEMORANDUM FOR: Michael R. Johnson, Chief
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FROM: Mark A. Cunningham, Chief /RA/
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SUBJECT: PROBABILISTIC RISK ANALYSIS PLAN FOR ADVANCED REACTORS

Enclosed is a copy of the "Plan for Probabilistic Risk Assessment Development and Application in Support of Advanced Reactors," dated September 13, 2002. This plan identifies the work that RES intends to perform in the area of probabilistic risk assessment (PRA) in anticipation of receiving applications for licensing of nuclear power plants with an advanced reactor design. This plan has been developed in conjunction with our contractor, SNL, and has been reviewed by all Divisions within the Office of Nuclear Regulatory Research. All comments from RES divisions and those applicable from NRR's review of the RES Advanced Reactor Plan have been incorporated into this version of the plan. The main comment from NRR on the PRA section of the RES Advanced Reactor Plan was for more detail; the attached PRA plan provides more details of RES's proposed work.

We are requesting comments in the area of PRA where: (1) we have identified work that NRR believes is not needed to support license application reviews for advanced reactors, and (2) where there is work that NRR believes is needed to support license application reviews for advanced reactors, but such work is not identified in the plan. Other comments are also welcome.

Please review this plan quickly to ascertain whether you would like to have a meeting to discuss one or more aspects of the plan. We would be willing to arrange a meeting of the interested parties to assist in your review. If such a meeting would be beneficial, or if you have any questions on the plan, please contact John N. Ridgely of my staff at 415-6555. We request your comments by October 18, 2002.

Enclosure: As Stated

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Plan for Probabilistic Risk Assessment Development and Application in Support of Advanced Reactors

September 25, 2002

Table of Contents

Tables

Figures

Plan for Probabilistic Risk Assessment Development and Application in Support of Advanced Reactors

1.0 INTRODUCTION

1.1 Background

The nuclear industry has been exploring new and innovative reactor design concepts that promise to simultaneously attain performance and economic improvements while preserving the defense-in-depth philosophy with higher levels of safety. New reactor designs being considered include high temperature gas cooled reactors (HTGRs), such as the Gas Turbine Modular Helium Reactor (GT-MHR); advanced light water reactors (ALWRs), such as European Simplified Boiling Water Reactor, (ESBWR), AP1000, and International Reactor Innovative and Secure (IRIS); and advanced CANDU reactor, such as the ACR-700. The existing Nuclear Regulatory Commission's (NRC) research and regulatory infrastructure supports the licensing of light water reactors (LWRs). Changes to the research and regulatory infrastructure may be beneficial for supporting the licensing of new reactor design concepts. The NRC has developed an advanced reactor research plan that will develop the expertise, tools, and methods needed to support the advanced reactor licensing process. The research plan will continue to evolve as information becomes available.

One of the key research activities within the scope of the advanced reactor research plan is the development of a risk-informed, performance-based, regulatory framework. (It should be noted that "risk" does not necessarily refer to any consideration of offsite consequences.) Risk-informed decision making needs to include probabilistic risk assessments (PRAs). PRA will be an integral part of an advanced reactor applicant's submittal and will play a significant role in risk-informing the licensing process for new reactor designs. Therefore, the NRC should be prepared with the PRA tools and expertise.

The NRC has performed several PRAs, and has promoted its use as a means of developing nuclear power plant risk perspectives and identifying improvements. As a result, the NRC has developed the capability to use PRAs in regulatory decision-making for operating reactors. This capability is founded on the staff s in-depth understanding of the techniques and data employed in a PRA, the design and physical characteristics of the reactors modeled, and how the design and characteristics are modeled in a PRA in terms of underlying hypotheses and data.

However, advanced reactors (especially PBMR, GT-MHR, IRIS, and ACR-700) are new designs and, therefore, the current PRA experience needs to be expanded. The limitations of the current PRA experience applies to system modeling approaches and associated underlying hypotheses (e.g., treatment of passive systems); to failure data; and to the design, materials, systems, and safety approach. These limitations should be addressed. The tools, expertise, and data (including information related to uncertainties) should be developed to enable the staff to evaluate advanced reactor PRAs.

An advanced reactor PRA interfaces with virtually every other area of the advanced reactor research plan. Given that performing a PRA is an iterative process, knowledge of reactor systems, fuels, materials, human performance, and instrumentation and controls (I&C) is needed for postulating accident initiators, modeling of systems, and quantifying accident sequences. The results of an advanced reactor PRA will indicate what issues are important from a risk perspective and what areas need further investigation.

Some of the characteristics of the advanced reactor designs are significantly different from currently licensed reactors. Some of these differences will require the Commission to make new policy decisions, and some require technical decisions. Some policy issues have already been identified to the Commission in a Commission Paper (SECY-02-0139) dated July 22, 2002. The policy issues include: implementation of the Commission's expectations for enhanced safety, identification of the specific attributes for defense-in-depth, the role of PRA in establishing the licensing basis, identifying when accident-specific source terms should be used, and the issue of containment versus confinement. Some key technical issues have been identified to the Commission in a memorandum from William D. Travers, Executive Director for Operations, dated July 22, 2002. The technical issues relate to fuel fabrication, testing, and analysis; material's capabilities at different operating conditions; reactor systems analysis capabilities; and the role of PRA in advanced reactors.

1.2 Objectives

The purpose of the advanced reactor PRA research effort is to develop the methods, expertise, and technical basis needed for risk-informed decisions and to provide support in the decision-making process of licensing advanced reactors.

In the past, the selection of licensing basis events was done on the basis of sound engineering judgement, and the approach to licensing was to provide safety margins and use the philosophy of defense-in-depth. However, experience has shown that this approach can lead to unnecessary conservatism and may not have adequately addressed all of the potential safety concerns that currently have been identified. PRA supplements the engineering approach and provides a tool to identify potential weaknesses in both design and operations, especially when used in an iterative manner.

The use of PRA in the licensing of advanced reactors is more challenging than the use of PRA in risk-informing current regulations. Applicants will provide technical arguments for the adequacy of their proposed advanced reactor design, in part, on the basis of PRA results. While safety margins and defense-in-depth will be maintained to protect the health and safety of the public, PRA results and insights may be used to enhance the traditional approach and to reduce the conservatism traditionally provided. Therefore, developing the PRA tools, methods, and expertise is important for the review and licensing of these reactors. Having this capability will enable the staff to do comparisons with submitted analyses and results, thus, gaining an independent and more complete understanding of the safety issues associated with the proposed designs. These tools, methods, and expertise are also needed for direction of other areas, e.g., identification of the most probable accident scenarios for accident modeling and source term identification with MELCOR and consequence assessment with MACCS2.

The objectives of the advanced reactor PRA work are to develop the guidance for NRC reviewers, explaining how the results can be used to independently review advanced reactor PRAs, and to support the development of a risk-informed regulatory framework. To obtain these objectives, it is necessary to develop: the necessary data for the PRA, an understanding of the uncertainties, the methods necessary to understand the PRA aspects of advanced reactor designs, the expertise to evaluate advanced reactor PRAs, and identification and prioritization of additional research needed.

1.3 Scope

The PRA research and development will be sufficient to address the cornerstones of reactor safety and cover all activities from the receipt of fresh fuel to the shipping of spent fuel. As such, this plan addresses the methods needed to generate full scope PRAs for the reactor and other sources of radionuclides, such as spent fuel and radioactive waste. The PRA methods will address internal and external events, and all risk significant operating modes. The methods needed to estimate the risk to both the public and the environment will be addressed. Dependent upon the risk metrics chosen for use in licensing advanced reactors, additional methods may be required to evaluate the risk to workers. The work addressed in this plan does not include NRC review of any applicant's PRA.

1.4 Planned Activities

This plan is comprised of three aspects. The first is to perform independent limited scope PRAs for advanced reactor designs and identify how the results will be used to support the licensing process. Specifically, the results will be used to: (1) gain expertise, (2) provide guidance for assessments in other areas of the advanced reactor research plan, and (3) evaluate advanced reactor designs. The second aspect, which is integrally related with the first and will be performed concurrently, is to develop any new PRA methods, data, and tools needed for evaluating the different design and operational characteristics of advanced reactors from those of current reactors. The third aspect is to document what has been done and to provide guidance for the review of applicants' advanced reactor PRAs.

This plan includes efforts to perform PRAs for advanced reactor designs and to identify how the PRA will be used to support the licensing process. Figure 1 illustrates how a reactor-specific PRA developed by the staff or its contractors will be used in this effort. The PRAs can be used to identify and prioritize research needs, to help evaluate an applicant's advanced reactor design, and to support the development of a risk-informed regulatory framework. The process illustrated in Figure 1 is iterative. The level of detail of the PRAs is expected to increase as the design progresses from conceptual to detailed. As such, the PRAs can serve as a risk management tool to help the staff better understand the design and develop needed regulatory decision-making criteria.

It is clear that future risk-informed approaches need to include a comprehensive treatment of risk, including all possible hazards and risk significant operating modes. The performance of PRAs will provide critical information for risk-informed regulation and will help identify where new PRA methods need to be developed. There are certain areas where PRA research is needed, regardless of the reactor technology chosen, and there may be design-specific areas requiring research. An examination of the current state of PRA capabilities and an identification of gaps will be an early, high priority item.

Examples of particular areas where PRA research will be needed include (not in any particular order):

- limited scope PRAs will be needed to identify issues and focus research,
- digital I&C reliability, including software reliability,
- improved treatment of uncertainties in decision-making,
- human reliability analysis, particularly considering extensive use of digital I&C in control rooms and other advanced equipment,
- dynamic models that deal with varying plant configuration,
- next-generation PRA software development,
- accidents involving radioactive materials outside the reactor core, and
- possibly security and non-proliferation issues.

Reactor license applications will include a PRA. This plan addresses the development of a Regulatory Guide that identifies what would be an acceptable submittal with regard to the PRA. In addition, the generation of guidance for the staff evaluation, i.e., Standard Review Plan sections, of a submitted PRA is also part of this plan. The development of a high-level document that identifies how to use the results of both applicants' and staff PRAs in regulatory decision making is also proposed. An offshoot of these efforts is the development of PRA methodology documents.

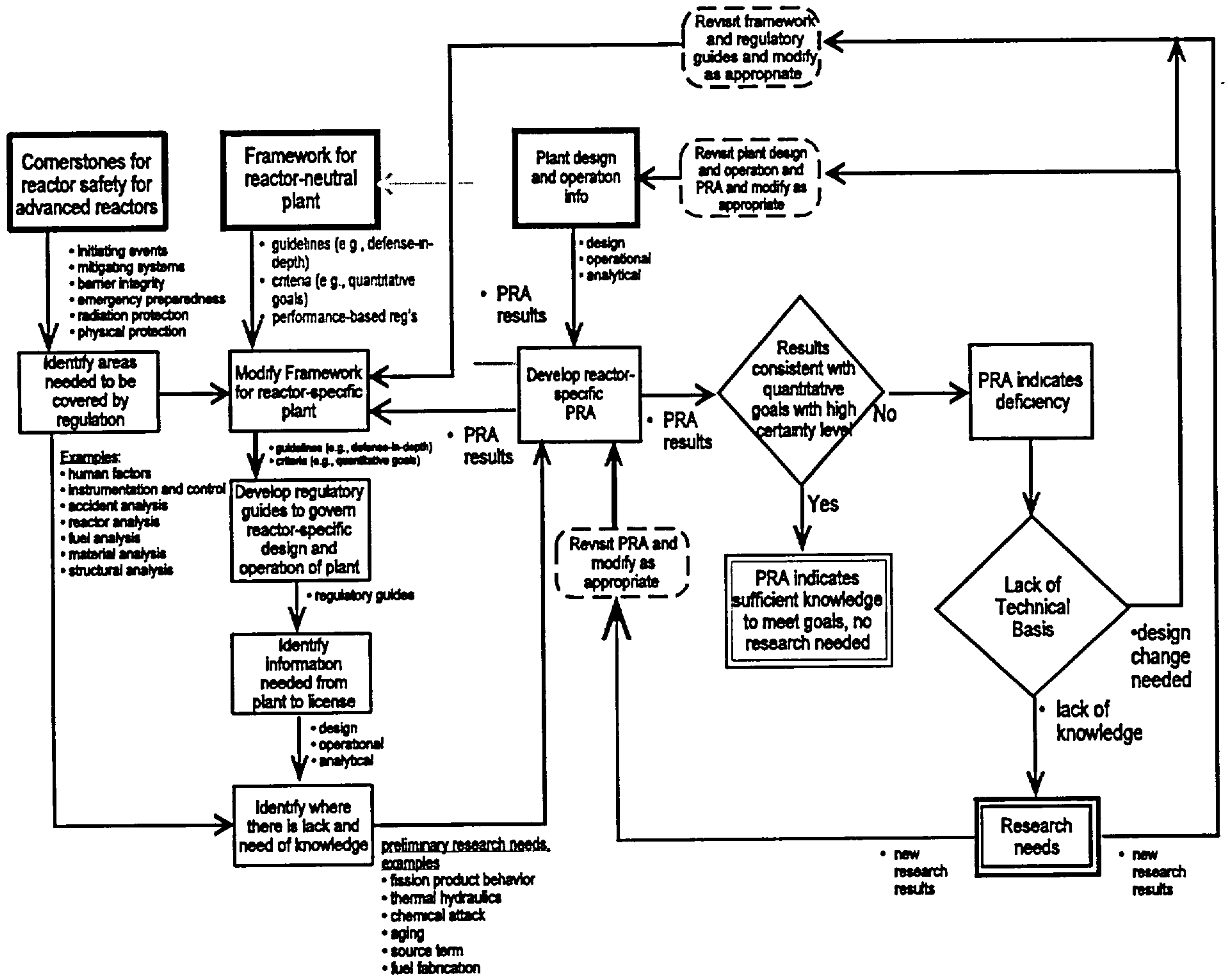

Figure 1. Use of PRAs to Support the Licensing Process

1.5 PRA Applications in Advanced Reactor Regulation

The role of PRA in the licensing of advanced reactors will be greater than for the current generation of LWRs since the PRA can be used throughout the design and licensing phases to identify risk-significant accidents and needed mitigating features. At the conceptual design level, preliminary (limited scope) PRAs can be used to evaluate the adequacy of the proposed design to respond to postulated accidents. Early interaction between the designers and PRA analysts can lead to a risk-informed design. The generation of PRAs will allow the identification of risk-significant accident scenarios and key uncertainties in the plant response. These key uncertainties indicate where research is needed. In addition, the results of preliminary PRAs can be used to help identify and provide technical support for design-specific regulatory guidelines.

As the design of an advance reactor progresses, additional insights into the adequacy of the design, the need for additional research, and the development of the regulatory basis will proceed. However, this is only possible if interactions between the designers and the NRC occur. It is likely that the evaluation of a design and subsequent interactions will not occur until a standard design certification is requested. Design information is needed to allow the NRC to review the Safety Analysis Report and PRA for the design. The review of the submitted PRA and possibly generation of confirmatory PRA models by the staff will identify if the design meets the Safety Goals and if additional research is needed to confirm the adequacy of the design. The PRA review process will be greatly enhanced by the staff developing an independent PRA.

A request for a standard design certification may not include site-specific considerations. Thus, only a general risk assessment of external events is possible until a combined license is requested by an applicant. A site/plant-specific PRA is required as part of a combined license submittal.

2.0 PRA DEVELOPMENT AND USE FOR ADVANCED REACTOR REGULATORY SUPPORT

Both industry and the NRC are using PRAs in the licensing process for advanced reactors. Unfortunately, there is little PRA experience with these new reactor types for either industry or the NRC staff. As a result, the development of PRAs by the staff is beneficial for the staff when reviewing the applicants' PRAs. The developed PRAs will provide the technical basis needed to assure that the submitted PRAs are both accurate and complete.

This section addresses the efforts to perform PRAs for advanced reactor designs and also identifies how the PRAs will support development of a risk-informed regulatory framework. This includes identification of research needs, development of risk-informed decision-making criteria, and actual evaluation of a design for comparison with the Safety Goals. The level of detail of the staff-generated PRAs is expected to increase as the design progresses from conceptual to detailed (assuming design information is available to the staff). Alternatively, it may be more cost effective for the staff to just review the detailed PRAs submitted as part of standard design certification request or a combined license submittal. In either case, the staff and applicant's PRAs can serve as a risk management tool to help the staff better understand the design and develop needed regulatory decision-making criteria.

2.1 Advanced Reactor PRA Development

Applications for new reactor licensees will include a PRA. The NRC will need to either review the applicant's PRA or perform an independent PRA. Developing independent limited scope PRAs is proposed for each advanced design based on similar designs or characteristics and with the available design specific information. In the course of performing these PRAs, the staff will explicitly gain an understanding of the new design, the contributors to its risk, and the uncertainties in that risk. The PRA development and analysis will help provide the staff with the expertise to review a PRA submitted by an applicant. Furthermore, the staff will be able to identify more completely additional areas where more information may be needed in order for a PRA to be adequately reviewed.

Limited scope PRAs, based on conceptual information, will be developed for advanced reactors as they come in for review. Each PRA will include the subtasks typically performed for a light water reactor (LWR): hazard identification; initiating event identification; accident progression analysis including success criteria, accident phenomena, and accident sequence delineation; systems analysis; data analysis; human reliability analysis; quantification and results analysis including uncertainties; and analysis of other operational states (e.g., low power).

Research and analysis efforts by both designers and the NRC will provide the specific data necessary to quantitatively support the risk-informed licensing of a reactor. Where applicable, existing information along with new research results needs to be reflected in the applicant's PRA and the results, including uncertainties, used to make corresponding licensing decisions. It may be necessary for the NRC to perform their own risk assessments of specific issues in order to make the necessary risk-informed regulatory decisions. At the extreme, the NRC may need to perform an independent PRA of the detailed design for each prototype. Whether or not to generate these detailed PRAs or rely on the review of submitted PRAs is identified as a policy issue in Section 6.

The regulatory decisions will likely be a function of the level of detail of the design. In the early stages of design, less design detail will be available and, thus, the staff evaluation should emphasize the major design features and regulatory issues first. When these major features and issues are resolved and when more design detail is developed, more detailed issues can be addressed.

Another output of the staff-generated PRAs will be the identification of areas where improvements in PRA methods and data are necessary to support the use of PRAs in the risk-informed regulatory process. The relative importance and uncertainty associated with the needed methods and data will be used to prioritize their development.

2.2 PRA Application for Advanced Reactor Regulatory Support

The results of the limited scope PRAs developed in Section 2.1 will be used to: (1) gain expertise, (2) provide guidance for assessments in other areas of this plan, and (3) develop an independent capability to evaluate advanced reactor PRAs. The level of detail in these analyses will be determined by the information needed from the PRA to support the licensing process and the availability of design information provided. The results will provide a basis for performing comparisons with the applicants' PRAs.

This section identifies how the PRAs of different levels and scopes developed and described in Section 2.1 will be used to support different regulatory activities. This includes identifying and prioritizing research needed to provide the technical support for licensing of advanced reactors, which would include the review or auditing of applicants' submitted PRAs. These activities are discussed in detail in the subsequent sections.

2.2.1 Use of PRAs to Identify and Prioritize Research

A strong research program is essential to the safe and efficient operation of the next generation of nuclear reactors. Research is necessary to ensure that the selected strategies and tactics are feasible and that the decision-making criteria are being met. Research should not be done for the sake of research, but rather to provide necessary technical support for the development of regulations and decision-making criteria. Particular projects should be clearly tied to these goals and have a schedule that supports the overall licensing activities.

Apart from the logical need for a research program to support advanced reactor regulation, the requirements of 10CFR52.47 are likely to stimulate significant research needs. This regulation requires a significant amount of analysis and testing over a wide range of normal and accident environments to validate new designs. Substantial research is needed to develop the tools to support and verify such analysis and the development of criteria for performing this analysis and testing.

While detailed plans for research projects will take more time to develop, it is possible to identify general areas where research will be needed and to prioritize them. PRAs generated by the staff (at the conceptual design level and possibly for the prototype plants) may assist in both identifying and prioritizing research needs. Identification of these areas is based on the premise that meeting the Safety Goals and providing defense-in-depth will be required. Research will provide the confirmation that the objectives are being met.

A variety of information is needed to provide inputs to risk models and to explore the behavior of systems under a variety of circumstances. In addition to particular deterministic calculations, there is a need to provide uncertainty information to risk models. To that end it is important to begin to take advantage of advanced computing and uncertainty analysis techniques. These techniques have been developed and implemented in the DOE community and elsewhere, but have received little use within the reactor safety community. Examples of areas where research is appropriate include:

- decay heat removal by conduction and radiation in the HTGR designs,
- impact of IRIS containment design on small/medium LOCAs,
- HTGR fuel performance, and
- aging affects.

The PRAs could identify accidents, systems, and the phenomena that contribute significantly to risk. For phenomena or areas where there is uncertainty that may affect the licensing of a design, there is a need to develop ways for using the PRA to support the identification and prioritization of regulatory research efforts. This may include sensitivity studies and use of importance measures to determine the risk potential resulting from the phenomena.

2.2.2 Use of PRAs to Support the Licensing Process

As illustrated in Figure 1, the PRA evaluations identified in Section 2.1 will be used to support the development of a regulatory framework including risk metrics. Here, the PRA will be used to support the technology-neutral regulations (e.g., the general design and operational criteria). Risk-informed alternatives to current requirements will be examined (e.g., replacement of current safe shutdown path approach used in Appendix R with a risk-informed approach and replacement of single failure criteria with reliability criteria). In addition, the PRA for each advanced reactor design will be used to help identify and to provide the technical support to develop design-specific regulations and Regulatory Guides.

Advanced reactor designs are striving to provide sufficient defense-in-depth through simpler technological solutions. Risk-informed decision making will play an important role in the development and optimization of future reactors to achieve higher levels of safety and to reduce the cost through simplification of safety systems. Risk-informed decision making will also contribute to a safety classification of structures, systems, and components (SSCs) based on a probabilistic ranking of their importance to safety. In this regard, specific areas to be explored in this task are how to use the design-specific PRAs to:

- identify design basis accidents (DBAs),
- special treatment requirements,
- categorization of equipment,
- support of inspection and monitoring programs,
- regulatory oversight of key assumptions, and
- identification of confirmatory research.

For example, the basis for event sequences controlling the design and special treatment requirements can include both deterministic design basis events and PRA results. In addition, expansion of the methods generated in the SECY-00-0198 Option 2 work for special treatment requirements and SSC categorization will be explored. For example, consideration will be given to using the Top Event Prevention Analysis method to support identifying the safety-related equipment in a design.

The concepts of core damage frequency (CDF) and large early release frequency (LERF) may not be the best figures of merit for advanced reactor designs. Therefore, the question of what are the appropriate subsidiary figures of merit for advanced reactor designs must be addressed. Identification of the appropriate risk figure(s) of merit is needed for risk evaluations as well as for developing regulatory criteria for design review and acceptance. Thus, this task will identify the appropriate risk metrics for advanced reactor designs. These risk metrics must be consistent with the NRC Safety Goals. In addition, metrics may be identified to address the risk to workers and the environment, which is a policy issue that needs to be resolved by the Commission. PRAs should be capable of calculating these metrics.

While much work remains to be done, expectations are that implementation of risk-informed methods will provide valuable insights for evaluating the design and resolving technical issues. For example, a very detailed evaluation of offsite release sequences (both their frequency and magnitude) could be used to provide the technical basis for the Commission to address the policy issues concerning the appropriate offsite emergency planning needed for advanced reactors.

During the process of reviewing an application for an advanced reactor license, there may be instances where a risk evaluation may be beneficial in determining the importance or in formulating a resolution to a newly discovered design, material, or operational issue. In addition, PRA evaluations will likely be an integral part of most of the application review effort. PRAs will assist in making the determination of a plant design meets the Safety Goals or the appropriate risk metrics. Finally, PRA insights will be used in developing the technical basis for the Commission's decisions on advanced reactor policy issues. These issues, as identified in SECY-02-0139 dated July 22, 2002, include: (1) implementation of "enhanced safety" over current operating plants, (2) the role of PRA in plant licensing, (3) usability of scenario-specific accident source terms for licensing decisions and site suitability, (4) use of a confinement instead of a containment, and (5) acceptability of reduced emergency planning zones because the new plants are "safer."

3.0 PRA METHODS DEVELOPMENT FOR ADVANCED REACTORS

Advanced reactors include technical features that eliminate some accident sequences by design. Examples of such features include passive systems, reduced power densities, increased design margins (e.g., use of the 1600°C fuel temperature limit in HTGRs), and increasing the time constants for the overall reactor system in order to slow down the transient response of the system. These measures should provide tolerance to failures, allowing more time for automatic control and operator actions to mitigate an accident. In addition, they simplify the design of control systems and the actions required from the operators, and decrease the number and severity of challenges to structures and systems. Such features will affect the development of PRAs for these advanced reactor designs.

This section provides the plans for identifying, prioritizing, and developing the methods, tools, and data needed for performing PRAs for advanced reactors and to transition to a risk-informed regulatory framework. Needed PRA methods, tools, and data will be identified based on the current state of PRA, the unique aspects of advanced reactor designs that need to be modeled, and the quality needed for risk-informed decision-making. The needed methods, tools, and data will be prioritized with regard to their potential impact on the risk assessments of the designs and the corresponding impact on licensing decisions.

3.1 Identify PRA Modeling, Tools, and Data Required

The objectives of this task are to identify any improvements in PRA methods, tools, and data necessary for obtaining PRA insights for advanced reactors to support licensing activities and development of risk-informed regulatory framework.

A preliminary review of existing PRA methods was performed to determine where existing methods may be insufficient for advanced reactors or for licensing applications. This initial review covered all of the technical elements of a PRA (i.e., initiating events, system modeling, accident sequence analysis, containment performance, external events, etc.) and some specific issues that may improve the evaluation of the risk from nuclear power plant operation. A review of the adequacy of existing PRA tools and what data to use for these PRAs was included. The review considered the reactor and other sources of radionuclides at a plant covered by regulations (e.g., spent fuel and radwaste), all risk-significant operating modes, and includes both internal and external events. A preliminary list of PRA research activities necessary to support advanced reactor regulation support is provided in Table 1. In some cases, there is an existing research program or a planned activity in the advanced reactor research plan that addresses the identified issue. Any additional work required to support advanced reactor PRAs is identified for these issues.

Table 1. List of advanced reactor PRA research activities.

PRA Element	Description of Research Activity
Fuel Quality and Performance	Develop acceptable goals and uncertainty values for quality of manufacture, handling, and monitoring
Initiating Event Identification and Quantification	Identify the initiating events and quantify their frequencies
Accident Sequence Analysis	Analyze accident sequences to identify core damage frequencies and large early release frequencies
System Modeling	Develop methods for modeling and quantifying the failure of passive systems
	Incorporate appropriate results from the efforts under Job Code Y6332[1] to develop methods for assessing the reliability of digital I&C systems
	Develop methods for modeling "smart" equipment in PRAs
Data Collection and Analyses	Identify and evaluate existing data sources for applicability to advanced reactors and determine appropriate methods to generate required data distributions
	Generate data (initiating event frequencies, component failure rates, common cause failure probabilities, and component fragilities to fire, seismic, and other harsh environments) for use in NRC PRAs
External Events	Evaluate effects of seismic events
Accident Progression and Containment Performance	Identify appropriate risk metrics
	Evaluate containment/confinement issues
PRA Quantification	Develop methods for addressing model and completeness uncertainty
	Incorporate appropriate results from SAPHIRE to quantify full scope PRAs (Job Code Y6394[2])

[1] "Digital Systems Risk"

[2] "Maintain and Support SAPHIRE Code & Library of PRA"

Table 1. List of advanced reactor PRA research activities.

PRA Element	Description of Research Activity
Risk Importance Measures	Review of alternative risk importance measures/tools for ranking equipment being performed under Job Code Y6547[3].
Dynamic Modeling	Identify the capability of dynamic modeling to support the advanced reactor regulatory process
	Determine the feasibility of developing dynamic PRA tools using existing computer codes
Human Reliability Analysis	Incorporate appropriate results from the development of data for use in HRA (Job Code Y6123[4])
	Review adequacy of existing HRA methods for evaluating operator response in advanced reactors considering new design factors such as multiple modules, digital I&C, and smart equipment
Safeguards and Security	Determine the use of PRA results for use in risk-informing safeguards and security requirements
	Identify the requirements necessary to expand PRA methods to evaluate sabotage

The identification of specific needs was based on the knowledge of advanced reactor designs (e.g., modeling digital I&C, data needs, and passive system modeling) and current PRA methods (e.g., model and parameter uncertainty, errors of commission, aging affects). Input from cognitive experts was used in this effort. Additional needs that may be identified during the course of performing PRAs for the designs will be addressed.

This plan will benefit from the input of experts in all areas of PRA. This includes recognized experts both inside and outside the NRC.

The methods review will also evaluate and identify the risk metrics that could be used for advanced reactors. Use of metrics other than CDF and LERF may require modification or development of additional PRA methods and tools. The PRA methods development will interface with the other areas in the RES advanced reactor research plan, particularly in severe accident and consequence areas, and advanced reactor licensing framework development.

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[3] "Improved Methods for Performing Important Analyses"

[4] "Retrospective Risk Assessment of Human Reliability in Operating Reactors"

A preliminary identification of potential PRA methods, tools, and data improvements is provided below.

Fuel Quality and Performance

Applicants for HTGRs place a greater emphasis on fuel performance than current LWR licensees. It is alleged that the HTGR fuel will not fail and release fission products into the coolant because of a combination of high integrity coating of the particles (and pebbles in the PBMR) and the alleged impossibility of attaining failure temperatures in the core. This argument is used to support the notion that containment is not needed, only a confinement may be necessary and defense-in-depth is achieved by taking credit for emergency planning. All of this lays the burden on the quality and performance of the fuel particles. Given that there will be on the order of 5×10^9 fuel particles in the reactor core, the question needs to be answered as to where and to what level of detail should the NRC be involved in the details of: (1) fuel fabrication, quality control, testing, and transport, (2) on-line refueling, and (3) storage of spent fuel of fuel particles/pebbles. Therefore, it is necessary to identify the performance goals required to support the applicant's claim of not needing a containment. These goals need to be for: (1) the quality of particle fabrication, (2) the uncertainty acceptable for a particle quality assurance testing program, (3) assurance for proper fuel loading in the core (or pebble distribution in a PBMR), (4) the acceptable uncertainty level in on-line fuel monitoring accuracy, and (5) fuel handling and storage accidents (applicable only for the PBMR). This information is needed to address the total plant risk as well as support the technical basis for resolution of the technical and policy issues by the Commission.

Initiating Event Identification and Quantification

The first step in assessing the risk associated with a given reactor design is to use a systematic approach to identify the events that can challenge the plant operation. The events that challenge the current generation of LWRs may not represent a complete set of challenges to advanced reactors. Those events that have the potential to initiate an accident should be completely and comprehensively identified. This can be accomplished with a top-down approach to first identify the top events, such as excessive power, excessive heat-up, etc., and then employ a systematic process to identify the events or conditions that could lead to the top or undesirable state. The process can help address completeness issues associated with the identification of initiating events for advanced designs. Searches for applicable events at similar plants (both those that have occurred and those that have been postulated) and by use of existing deductive methods (e.g., top logic models, fault trees, and Failure Modes and Effects Analysis) will provide valuable insights. Extensive use of existing HTGR design and PRA information will be used, as appropriate for the GT-MHR. Similarly, LWR experience and data will be used as appropriate for IRIS.

The need to develop additional methods for identifying initiating events is not anticipated. However, the unique characteristics of the advanced reactor designs will require application of existing methods to these unique aspects. For example, an advanced reactor design that uses continuous refueling presents the potential for accidents associated with these refueling efforts.

The HTGR primary coolant pressure boundary (PCPB), including the "vessel" between the reactor and the generator, will need to be carefully considered for two reasons. First, the PCPB may experience greater challenges (higher temperature excursions) during severe accidents than current designs. This is a materials issue, and the PRA will need to interface with the material area of the RES advanced reactor plan. The second reason is that there will be multiple reactor modules on a given site. The effect on a PRA of multiple modules on a site needs to be evaluated.

Since there will be no operational experience for these reactor designs for some time, the frequency of accident initiating events modeled in the PRA will initially have to be estimated. Existing data from available sources can be used in this effort. The effort to identify and address the applicability of available data sources for use in advance reactor PRAs is discussed below under "Data Collection and Analyses."

Accident Sequence Analysis

Current PRAs are usually performed for a single unit or sometimes for two sister units. Advanced reactors (e.g., GT-MHR) may operate multiple modular units together at a site with a centralized control room. The PRAs for modular reactor designs need to address potential interactions among the multiple units. This includes common accident initiators, common support system dependencies, interactions between units caused by accident phenomena (e.g., smoke generated by fire), and the potential effects of smaller operator staffs in a common control room under potential common cause initiators (such as seismic events). Additional PRA tools are not currently envisioned for modeling the accident behavior of multiple units (other than in the human response area which is addressed under "Human Reliability Analysis").

System Modeling

Advanced reactor designs are moving towards the simplification of plant systems with extensive use of passive features. A simplified system is one that is more easily operated and maintained or has reduced the number of components necessary to provide the safety and performance functions (thereby reducing the number of failure points and modes) and, therefore, should be more resistant to human errors. Passive systems that rely on pressure, gravity, or thermal gradients offer the opportunity to reduce the number or complexity of active systems and potentially the need to rely on active safety-grade support systems. The challenge is to demonstrate the capability and reliability of passive systems to meet the core cooling requirements and to deal with their longer response time in PRAs. There is the potential for events to adversely effect the structural integrity of the passive systems, e.g., jet impingement could result in a failure of an accumulator support, causing the accumulator to fall and fail. This type of issue also needs to be considered. The approach used for the AP600/AP1000 design certification to treat passive system components will be considered as the starting point for treating passive systems in HTGRs.

The risk from reactors incorporating passive components is determined by initiating events of very low probability. The consequences of these events are determined by the direct phenomenological response of the plant to these events, rather than by a sequence of failures of systems, which individually have higher probabilities and can be analyzed with much less uncertainty. An example of this is the reliance on passive heat conduction in HTGR designs to prevent exceeding a maximum fuel temperature of 1600°C. Methods for dealing with the uncertainty in phenomenological responses such as this without relying solely on the traditional conservative approach (e.g., safety margins) needs to be developed.

Digital systems typically have not been used extensively in LWRs and, thus, have not been considered in many existing PRAs. In advanced reactors, I&C systems will normally be digital. Digital I&C systems may have different operational and reliability characteristics than the analog systems used in current LWRs. Thus, digital systems may have failure modes that are different from those in analog systems. For example, digital systems may fail due to smaller voltage spikes or sooner under loss of cabinet ventilation. Inadequate consideration of potential digital system failure modes can lead to the failure of the system to function properly under postulated conditions.

It is not readily apparent that these reliability aspects of digital systems can be addressed with existing PRA methods. As a result, the reliability of digital systems is currently being addressed in existing research efforts (Job Code Y6332). This research effort will evaluate the digital systems analysis and tools under development and determine their strengths and weaknesses with respect to integration into PRAs. Guidance for including digital systems in PRA will be developed. Any additional research efforts needed to address the modeling of digital system reliability will likely be identified as a result of this effort. Thus, this work needs to be monitored. However, at this time there is no near term funding; this will be a limitation of the PRAs in the near term.

Automated surveillance and diagnostic systems, as well as artificial intelligence systems are currently being developed and likely will be incorporated in advance reactor designs within the next 10 years. The U.S. Department of Energy's Nuclear Energy Research Initiative (NERI) Program is currently developing a set of tools and methodologies designed to improve the reliability and safety of advanced reactors through the introduction of "smart" equipment and predictive maintenance technology. Smart equipment incorporates sensors, data transmission devices, computer hardware and software, and human-machine interface devices that continuously monitor and predict the system performance and remaining useful life of equipment. The use of smart equipment could replace the current practice of scheduled inspection and maintenance with maintenance or replacement dictated by the measured condition of the equipment and predictions of its continued performance. Underlying this approach is the need to understand how a history of sensor information (e.g., from a vibration sensor) relates to component's wear and tear and remaining useful lifetime.

The modeling of smart systems in PRAs needs to be addressed. The modeling considerations includes the reliability of the smart equipment including the sensors, data transmission devices, and computer systems (addressed under "Data Collection and Analysis"). In addition, the reliability of the software developed to predict the continued performance of equipment and the decision making process (i.e., artificial intelligence logic) should be addressed. Modeling of the operator response to the synthesis of the data provided by artificial intelligence logic and the contribution to human errors need to be reviewed (addressed under "Human Reliability Analysis").

Data Collection and Analyses

The use of appropriate data is crucial to the quality of the PRA. Advanced reactors introduce different systems and components and, hence, the data may not be sufficient and in some areas appropriate. Therefore, a research task to identify what type and quality of data are needed for advanced reactor PRAs is required. This task will also collect and analyze data applicable to advanced reactors during all modes of operation. This includes initiating event frequencies, component failure rates, maintenance unavailability, and common cause failure probabilities. In particular, data is needed for digital systems, smart equipment, and gas (i.e., helium) components such as compressors and blowers, The susceptibility of these components to failure in the environments created during accidents, including external events, needs to be addressed.

Existing databases and published sources of reliability data need to be identified and evaluated for applicability to advanced reactors. This includes data used in an existing NRC-reviewed HTGR PRA, foreign data for similar reactor designs, and data identified in the NERI program. Characteristics of each data source (e.g., generated from raw data versus expert elicitation, reactor versus non-reactor related data), the amount of data represented, and the characteristics of the data (i.e., failure modes, represented environments, component quality level, and applicability to advanced nuclear power plants) need to be carefully evaluated.

This task also will address data uncertainties. Understanding the uncertainties is a very important aspect for any PRA; this is especially true for advanced reactors, given the limited or lack of operating experience and the expected significant use of the PRA in the licensing process.

External Events

Current plants have shown that the contribution to core damage frequency from external events (defined here to include internal fires and external events such as earthquakes and external floods) may be similar to that from internal events. There is an existing research program to develop the technical tools and data to perform fire risk assessments (Job Code Y6037). It is not anticipated that additional or alternative methods will be needed for evaluating the fire risk for advanced reactors. Similarly, development of new methods to evaluate external events are currently not needed. However, the response to various external events should be evaluated because of the unique characteristics of advanced reactor designs. For example, the response of digital systems, such as computers and micro-processors in a fire or flood, may be different than that of electro-mechanical components. The differences may not be just in probability of failure but also in the failure modes and timing that could potentially occur. There is also the possibility of a reactivity insertion accident either during or as the result of a seismic event. While this may be of particular importance for pebble bed modular reactor designs, it may be applicable to other designs, such as the ACR-700.

Accident Progression and Containment Performance (including source term)

The concepts of CDF and LERF may not be the best figures of merit for some advanced reactor designs. Dose limits have been suggested as criteria for the HTGRs. A full Level 3 PRA would be required to evaluate this figure of merit. Other figures of merit may require just a Level 2 PRA. Additional figures of merit for worker risk from accidents may impose additional accident sequence considerations and require additional consequence tools. However, until these figures of merit are identified, no additional PRA tool or method development is proposed.

Success criteria, accident progression, and source terms for advanced reactors may be different from those for LWRs and should be understood. The likely accident progression phenomena need to be identified based on ongoing research, previous experiments, experience in other industries, and expert judgment. For example, although hydrogen should not be generated in HTGRs during the course of an accident, the potential for generation of other combustible gases needs to be evaluated. In addition, the loss of helium and the effects of air (and potentially water) ingress on the accident progression needs to be considered. A combined deterministic/probabilistic approach, with elicitation methods similar to those used for the liner melt through and direct containment heating issues in some LWRs, may be possible. Assessment of potential combustible gas generation, for example, will be performed as part of thermal hydraulics and severe accident work of the advanced reactor research plan and will be incorporated into the PRA as part of the necessary data to evaluate advanced reactors. Analysis of offsite consequence will be based on the severe accident and consequence work in the RES Advanced Reactor Research Plan. This work will support the technical basis for proposed risk metrics.

One of the policy issues that needs direction from the Commission is the possibility of permitting a confinement instead of a containment. In a Commission paper (SECY-93-092) the staff proposed an approach for containment that focused on functional performance, rather than prescriptive design criteria. The Commission, in a July 30, 1993, staff requirements memorandum (SRM) approved the staff's proposal. Because the new advanced reactor designs are different, this question has again been raised in a recent Commission paper (SECY-02-0139 dated July 22, 2002). A probabilistic containment analysis (Level 2 PRA) is needed to assess the benefits of having a reactor containment versus a reactor confinement with a filtered venting system for providing protection against release of fission products. (The confinement concept has been successfully modeled in past PRAs, although not for commercial reactor designs.) The technical assessment of the performance of containment versus confinement should be performed as part of the severe accident research work to support the technical basis for the Commission's policy issue decision.

The benefit of complete underground siting, instead of the partial underground siting now proposed for some HTGR designs, also needs to be evaluated. The results of the assessment will be used as input to the PRA models for the advanced reactors discussed in Section 2.1. These analyses could be applicable to the review of safeguard and security portions of license applications.

Source term work will be performed as part of thermal hydraulics and severe accident research work. The knowledge of fuel performance is a prerequisite to performing an independent review of the PRA. We need to understand how the core behaves in accidents such as overheating or immersion in media other than helium (i.e., air or water). This behavior should be understood not only for fresh fuel but also for end of life fuel to evaluate the impact, if any, of burn-up. This work, which will be performed as part of fuel performance assessment, thermal hydraulics, and severe accident work in the RES Advanced Reactor Research Plan, will be incorporated into the PRA.

PRA Quantification

Identification and quantification of uncertainties in a reactor PRA will help decision makers determine whether reducing the uncertainties by performing more research or strengthening the regulatory requirements and oversight (e.g., defense-in-depth and safety margins) should be pursued. A PRA provides a structured approach for identifying the uncertainties associated with modeling and estimating risk. There are three types of uncertainty: modeling, data, and completeness. The quantification of parameter uncertainty is well understood, and additional research is not needed beyond establishing those uncertainties (see the previous discussion on data). However, additional research is needed on how to formally address model and completeness uncertainty in a PRA and how to integrate the uncertainty assessment in a risk-informed regulatory process along with appropriate safety margins and defense-in-depth. In addition, the integration of uncertainty assessments in different parts of a PRA needs to be addressed. The use of dynamic modeling (discussed previously) provides one means to accomplish this.

The SAPHIRE code could be used in the performance of an independent PRA by the NRC staff. The code is currently being used by the staff for performing or reviewing conventional PRAs. The use of the code continues to reveal errors indicating that a considerable debugging effort is still required to bring the code to the needed level of maturity for staff and public use. Responding to the needs of other programs (e.g., the pressurized thermal shock rule), SAPHIRE capabilities are also being expanded to allow lower truncation limits, better integration of Level I and Level 2 analysis, and integration of different types of analyses. Experience shows that needs for improvements always emerge from new applications. Development of the tools and data for advanced reactors may identify beneficial modifications. As part of the dynamic modeling, the need for creating a new computer code will be assessed as compared to linking SAPHIRE with other codes (see "Dynamic Modeling").

The PRAs for the advanced reactors should include an assessment of low power/shutdown modes. These operational modes should be included in the selection of DBAs and the categorization of structures, systems, and components (SSCs). In addition, advanced reactors operating in other than full power mode, can include unique operating characteristics that need to be examined in order to develop an understanding of the total plant risk.

Additional PRA methods do not appear necessary to evaluate shutdown risk. However, consideration of low/power shutdown PRA should be included in the research tasks identified in this plan. For example, the evaluation of data needs to include low power shutdown initiator frequencies, maintenance frequencies and durations, and common cause events.

Risk Importance Measures

The results of PRAs can provide input to identify risk-important SSCs. Selection of risk-significant SSCs will be an important aspect of the licensing of advanced reactors. Risk-informed classification processes have been developed to support the Maintenance Rule, graded quality assurance, and special treatment requirements. All of these methods rely on the use of importance measures, the most common being the Fussell-Vesely and Risk Achievement Worth measures. The limitations of these measures and the potential use of additional importance measures (e.g., partial derivatives) and alternative methods (e.g., the Top Event Prevention methodology) to identify potentially important SSCs is currently being addressed by RES (Job Code Y6547). The need for additional research in this area will be determined as an outcome of this work.

Dynamic Modeling

Transient events in nuclear power plants involve interactions among the reactor core, operating and standby systems, and the operator. The physics of the transient response include reactor neutronics, fluid mechanics, heat transfer, and fission product transport. Changes in the process parameters during a transient can result in the change in a component's state, including starting and stopping the component's operation. Existing PRA models do not reflect the dynamic response of a plant. Current PRA frameworks are quasi-static where accident scenarios are represented by a sequence of systems failures or successes and human actions. Each sequence represents a snapshot in time where the timing of events is often selected to provide conservative results especially with regard to the operator response. The status of the plant systems (either failed or operating) help in determining the values of critical plant parameters (e.g., pressure, temperature, and water level) and the timing of critical events.

Realistic plant PRAs should include the treatment of the dynamic interactions among the plant systems, the I&C, and the operator. This would preclude the need to define the time when component failures occur (i.e., the time of failure can be treated stochastically) and allow modeling of degraded component operation. More realistic modeling of component operation in turn will allow a more realistic calculation of the plant parameters and the timing of critical events. In turn, the operator response can be tracked through a sequence including the feedback on system operation and the potential for human errors.

Since dynamic PRA modeling can provide the benefit of integrating the entire plant response and consequences to an accident into one model, an integrated assessment of uncertainty in the model parameters, the thermal-hydraulic response, and the operator response is possible. This may be particularly appropriate for advanced reactors that use digital I&C and passive systems for accident mitigation. It should be noted there is no funding for the evaluation of digital I&C for FY03, and only limited the funding for FY04 and FY05. Therefore, there is likely to be little improvement in the PRA related to potential effects from digital I&C failures in the near term.

Dynamic PRA modeling has seen limited applications, primarily due to computer code limitations. However, with the advances in computers, such limitations are significantly reduced. A research task is proposed to examine the use of dynamic PRA modeling as an audit tool for regulatory decision making. Such a tool would provide a better estimate of the risk uncertainty interval and, thus, allow for a more robust evaluation of a design. The research effort should explore the capability of existing dynamic PRA tools and the potential need for generating new tools (e.g., by linking SAPHIRE with MELCOR). The modeling of the operator response in a dynamic PRA framework also would be an area that would have to be addressed (see "Human Reliability Analysis"). In addition, work will be needed to establish guidelines for importance sampling (i.e., guidance on what to choose to sample).

Human Reliability Analysis

The operators' role in new reactors will be different than that in current generation reactors. Advanced reactors are proposed to be built on the premise that they will be human-error free and that, if an event occurs, human intervention will not be necessary for an extended period of time. Thus, the tasks to be performed by operating crews in advanced reactors will be different from that in existing control rooms. The operators' interactions with plant systems may be different in a digital I&C environment. Differences in the man machine interface related to new types of displays, touch screen controls, etc. may impact the potential operator errors. In the extreme, with "smart" control systems, the operators' role could become more of a "supervisory" task as opposed to the "hands-on" operation in current plants. Thus, the main "job" of the operators may be to monitor system behavior and ensure that shutdown occurs properly when necessary.

Although human factors and human reliability analysis (HRA) are related, issues related to human performance, such as the need for operator performance (e.g., staffing and training) are part of a different activity in the advanced reactor research plan. With regard to HRA for advanced reactors, second generation HRA methods (i.e., ATHEANA) will be adequate to address the modeling of operator actions. However, continued development of the methods is needed (Job Code Y6123), particularly in the development of formal methods to quantify the probability of unsafe actions. Appropriate improvements in the development of data to support the quantification of human error would be incorporated into the advanced reactor PRAs.

When dealing with long-term and slowly evolving accidents, such as those expected to be dominant in advanced designs with graphite-moderated reactors, special consideration may be needed in calculating the core damage frequencies. In addition, operator performance may be affected by having multiple modules that share the same control room. The likelihood of errors of commission or omission need to be understood under these conditions. Finally, the use of digital systems and smart equipment will affect the operators' role in running the plant and responding to accidents. A specific task is proposed to determine if (and what) modifications are warranted to appropriately incorporate the impact of such factors on the human performance in advanced reactors. There should be close integration of this HRA task with the human factors tasks.

As part of the effort to review the utility in dynamic modeling of accident scenarios, a review of the adequacy of HRA methods is also required. In particular, the uncertainty in the thermal hydraulic response of a plant can lead to different operator responses including errors of commission. A dynamic PRA framework will also allow an assessment of each of the operators' actions in the context of the previous actions. The various operator responses (including errors of commission or omission) can in turn affect the accident progression or mitigation. Whether existing HRA methods are adequate for incorporation into a dynamic modeling process or whether other models are required (e.g., full simulation of the operator response from situation assessment to response selection based on procedures and training) needs to be examined.

It should be noted that the current RES budget plan does not include advanced reactor funding for human factors through FY05 and has only limited funding for digital I&C work. Therefore, there is likely to be little improvement in the PRA related to potential effects from human factors and digital I&C failures in the near term.

Safeguards and Security

There are some portions of this work where explicit information can be generated regarding the safeguards and security for advanced reactor designs. The NRC should explore how this can be accomplished in the most efficient manner and what other areas the PRA studies can assist in this endeavor. In addition, the possible of expansion of PRA to evaluate the risk from sabotage is a policy issue discussed in Section 6. Sabotage may dominate the risk profile for advanced reactors that are inherently safe. A task to identify the requirements necessary to expand PRA methods to evaluate sabotage events is proposed as part of this plan.

3.2 Prioritize PRA Methods, Tools, and Data Development Needs

Faced with the reality of limited resources, there is a need to prioritize the PRA methods, tools, and data development requirements needed for advanced reactor designs. A prioritization could allow focusing limited resources on PRA methods development that are most important to the licensing PRA applications, including those areas that are most likely to reduce uncertainty in the results.

The NRC uses the agency's strategic plan performance goals to prioritize RES activities as part of the budget process. As such, staff activities could be prioritized as they relate to:

- maintaining safety;
- improving effectiveness, efficiency, and realism;
- reducing unnecessary regulatory burden; and
- increasing public confidence.

This approach to prioritization would develop score ranging from 1 - 10, with 10 indicating the highest priority. Because the scoring system is not intended to numerically order the activities, it is important to note that more than one activity may have the same score.

This prioritization scheme was used to develop a preliminary prioritization of the PRA methods development work needed to support advanced reactor regulation. PRA development activities that have the potential to significantly contribute to the understanding of risk can be interpreted as improving realism, and increasing public confidence. Those activities that have the potential to reduce uncertainty in the risk evaluation can improve realism in the regulatory process and improve public confidence. If the development of PRA methods, tools, or data can resolve design or operational issues, there is a potential to reduce regulatory burden.

Application of the proposed prioritization method to the activities identified in Section 3.1 is summarized in Table 2.

3.3 Develop Methods, Tools, and Data for Advanced Reactor PRAs

Based on the identification of needed PRA methods development and the subsequent prioritization, the methods, tools, and data needed to support the use of PRA for advanced reactors will be developed. The PRA methods development task will help identify the guidance necessary for applicants to perform PRAs for advanced reactors and for NRC staff to review them. In addition, the results of these efforts will provide input into the documentation tasks identified in Section 4. Specifically, the results of the PRA development efforts will provide the technical support for the development of a PRA methodology document, Regulatory Guides, Standard Review Plan sections, and guidance for reviewing PRA submittals.

Each PRA methods development task should identify potential alternatives with selection of a method that provides sufficient results with minimal complexity. Potential use and integration with existing or past research programs in all areas (materials, aging, etc.) should be examined. Alternate or better PRA methods may be identified during the course of actually performing PRAs (by applicants or the NRC) for the reactor designs.

Specific regulatory issues can affect the PRA methods development effort. For example, a decision to generate risk-informed safeguards and security regulations may require development of PRA methods and data to perform such analyses. In addition, decisions to include risk-informed regulations for other sources of radionuclides outside of the reactor (i.e., spent fuel and radwaste) may also necessitate additional PRA methods development.

Table 2. Preliminary prioritization of advanced reactor PRA research activities.

PRA Element	Description of Research Activity	Maintain Safety	Improve Effectiveness, Efficiency, & Realism	Reduce Unnecessary Regulatory Burden	Increase Public Confidence	Priority
Fuel Quality and Performance	Develop acceptable goals and uncertainty values for quality of manufacture, handling, & monitoring	Y	Y	Y	Y	10
Initiating Events	Identify the initiating events and quantify their frequencies		Y		Y	5
Analyze Accident Sequences	Analyze accident sequences to identify the core damage frequencies and large early release frequencies		Y		Y	5
System Modeling	Develop methods for modeling and quantifying the failure of passive systems	Y	Y	Y	Y	10
	Incorporate appropriate results from the efforts under Job Code Y6332 to develop methods for assessing the reliability of digital I&C systems		Y		Y	5
	Develop methods for modeling "smart" equipment in PRAs		Y		Y	5
Data Collection and Analyses	Identify and evaluate existing data sources for applicability to advanced reactors and determine appropriate methods to generate required data distributions		Y		Y	5
	Generate data (component failure rates, common cause failure probabilities, and component fragilities to fire, seismic, and other harsh environments) for use in NRC PRAs		Y		Y	5
External Events	Evaluate effects of seismic events	Y	Y		Y	7.5
Accident Progression and Containment Performance	Identify appropriate risk metrics	Y		Y	Y	7.5
	Evaluate containment/confinement issues	Y		Y	Y	7.5
PRA Quantification	Develop methods for addressing model and completeness uncertainty	Y	Y	Y	Y	10
	Incorporate appropriate results from SAPHIRE to quantify full scope PRAs (Job Code Y6394)		Y			2.5
Risk Importance Measures	Review of alternative risk importance measures/tools for ranking equipment being performed under Job Code Y6547	Y	Y	Y	Y	10
Dynamic Modeling	Identify the capability of dynamic modeling to support the advanced reactor regulatory process		Y		Y	5
	Determine the feasibility of developing dynamic PRA tools using existing computer codes		Y		Y	5

Table 2. Preliminary prioritization of advanced reactor PRA research activities.

PRA Element	Description of Research Activity	Maintain Safety	Improve Effectiveness, Efficiency, & Realism	Reduce Unnecessary Regulatory Burden	Increase Public Confidence	Priority
Human Reliability Analysis	Incorporate appropriate results from the development of data for use in HRA (Job Code Y6123)		Y		Y	5
	Review adequacy of existing HRA methods for evaluating operator response considering multiple modules, digital I&C, and smart equipment		Y		Y	5
Safeguards and Security	Determine the use of PRA results for use in risk-informing safeguards and security requirements		Y		Y	5
	Identify the requirements necessary to expand PRA methods to evaluate sabotage		Y		Y	5

4.0 DOCUMENTATION AND PRA GUIDANCE

The manner in which risk-informed regulation will be implemented for advanced reactors may be different from current experience. A sound engineering approach to licensing will be implemented in conjunction with PRA as an integrated evaluation method. It is important to note that PRA should be applied to both the design and regulatory functions. It is expected that a designer will develop a design using risk-informed methods and then submit the design for regulatory review, during which regulatory issues are resolved which could result in some design changes. While some basic design and safety criteria may be established at the beginning of the process, most of the detailed design and regulatory criteria would be established during the actual design and regulatory review work.

In a risk-informed process, all significant design features and deterministic analysis results are reflected in the PRA. This means that the designer must iterate between deterministic design and evaluation tasks and PRA evaluations early in the design process. This means that probabilistic design criteria need to be established. Any design margin and subjective judgments made by the designer need to be incorporated into the PRA and corresponding design and safety decisions (e.g., defense-in-depth) made considering the PRA results.

This section addresses the plans for the development of a Regulatory Guide that addresses the adequacy of an advanced reactor PRA and the guidance for the staff evaluation of a submitted PRA. The plan also includes the development of a high-level document that identifies how to use the results of advanced reactor PRAs. Thus, the objective of this work is the development of PRA methodology documents.

4.1 Development of PRA Regulatory Guide and Review Guidance

PRA review guidance is needed to help the staff determine the probabilistic implications of different design configurations and operation conditions. The guidance will identify the perspectives needed to support NRC risk-informed decision-making throughout an advanced reactor licensing process. Such guidance must provide the staff with the knowledge that is necessary to understand both the results of a submitted PRA as well as the underlying hypotheses driving the results. Therefore, guidance will be developed to: provide staff guidance explaining how the results of this work can be used to independently review an advanced reactor PRA, interface and interact with the work performed in other areas of the advanced reactor research plan to feed its results back to help identify where there is inadequate information, and, thus, support research prioritization, and provide input to potential modification to the regulations and the development of regulatory guides and SRP sections.

4.2 Application and Use of PRA Results

Section 2.2 of this plan addresses the development of high-level guidance for using PRA results to support advanced reactor regulation. Specific applications of risk metrics and importance measures are addressed. This includes the use of PRA to support the NRC's efforts to potentially generate new regulations and to review applicants' submittals.

4.3 PRA Methodology Document

The development of new PRA methods explicitly for advanced reactor designs and to address other sources of radionuclides (e.g., spent fuel) and other types of accidents should be documented in a detailed PRA methodology document. Such a document would be complimentary to the Regulatory Guide and Standard Review Plan efforts mentioned above. Development of such guidance is part of this plan.

5.0 INTERFACES WITH OTHER ADVANCED REACTOR RESEARCH PLAN EFFORTS

This section addresses the interfaces between the PRA research plan and other parts of the advanced reactor research plan. As mentioned previously, PRAs will be used to help identify and prioritize research efforts in areas such as fuels, materials, structural, waste, and safeguards research. In addition, the interface with the efforts to establish a new regulatory framework will be identified. Interfaces with applicant and other stakeholder efforts will also have to be identified.

The focus of the advanced reactor research plan is to determine the critical information that will be needed to establish the safety standards for advanced reactors, explore the issues that involve significant safety uncertainty, and develop the analytical capability necessary to independently review these new designs. PRA will be used to help identify and prioritize the research that will be necessary to evaluate risk-significant accident scenarios and provide technical support for any necessary risk-informed changes to the regulations. In turn, other advanced reactor research efforts will provide information that will be incorporated into the PRA models. This integrated and iterative process where all technical disciplines work together is depicted in Figure 2.

The integrated process is described in the draft advanced reactor research plan. The following flow of information between technical groups as described in the research plan, would support the regulatory decision-making process.

Information in the form of data and analytical results generated by the fuels, materials, and structural technical groups provides key inputs into the reactor/plant systems analysis. In turn, reactor/plant analysis provides key information on plant operating and accident conditions back to the fuels, materials, and structural analyses technical groups.

Insights and data generated by the reactor/plant analysis (e.g., success criteria), together with performance information involving human factors considerations, I&C, and modeling assumptions, are used in the PRA and associated accident analysis activities. Accident analysis research identifies accident scenarios and frequencies for further and more detailed reactor system analysis and consequence assessment.

Insights from the accident and consequences analyses is then used to provide the technical support for the potential development of modifications to the regulations and the associated framework.

Identification of key accident scenarios is an important aspect of the licensing process. These events generally drive the regulatory decision-making process because they impact the classification of SSCs, and the plant design has to meet the requirements as related to potential offsite consequences. Thus, the accident analysis, consequence assessment, and regulatory decision-making processes all are interrelated. When significant accident scenarios for a plant design are identified from the PRA, reactor/plant analysis can be performed and the results used to place performance limits on the reactor fuel, reactor internals, and the safety-related structures, systems, and components. Additionally, reactor/plant analysis and associated sensitivity studies can be used to assess margins and develop PRA insights, which are crucial to the robustness of the accident analysis. The PRA provides perspectives that will support the regulatory decision-making activities and identify the research that is needed to support those decisions.

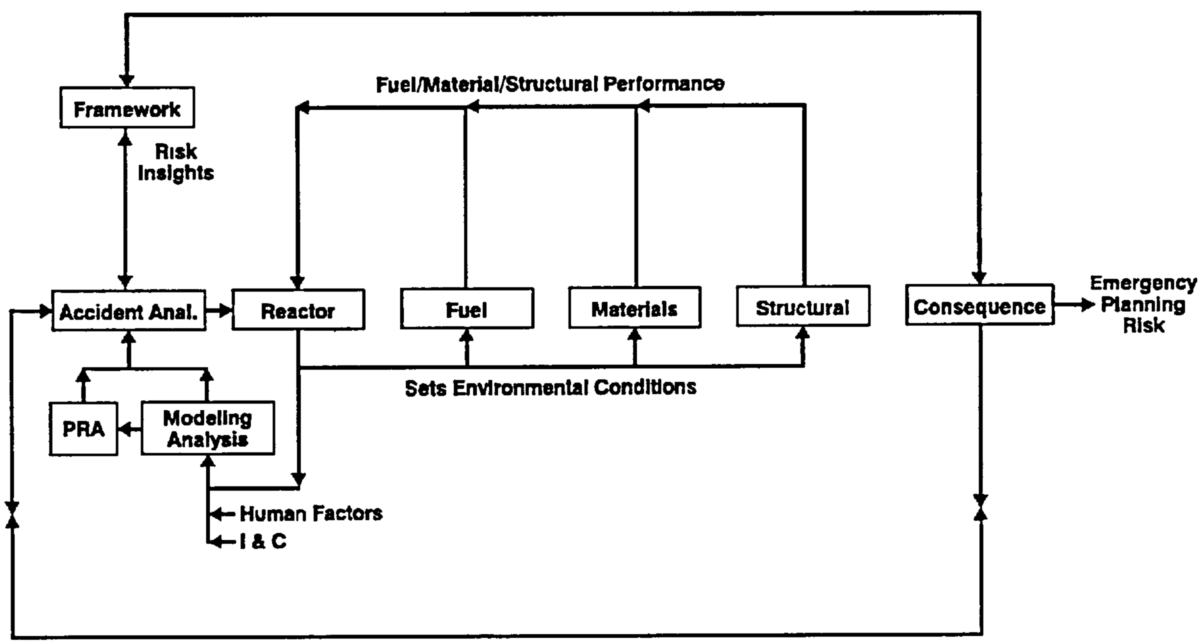

Figure 2. Information Flow Between Technical Areas

6.0 TECHNICAL AND POLICY ISSUES

Some of the PRA research efforts identified in this plan depend on the resolution of some technical and policy issues with regard to the scope of the PRA and its applications in the development of an advanced reactor framework. Currently identified policy issues are whether to have the NRC develop detailed PRAs for each advanced reactor design, to include spent fuel and radwaste in the scope of the PRAs performed by the staff (see Section 2), to use risk metrics other than CDF and LERF, and to develop methods for modeling sabotage events and evaluating the consequences to workers. These issues are discussed below.

In Section 2.1, it is proposed that the staff generate limited scope PRAs based on the design information available for each advanced reactor design as part of the pre-application review process. These limited scope PRAs would capitalize on previously performed PRAs of similar designs or characteristics with consideration of the differences and limitations. These PRAs will help identify the important accident scenarios, important phenomena that may require research, the importance of specific SSCs, help develop PRA review and submittal guidance, help in the, generation of the regulations and regulatory guides required for the licensing of the design, and provide the staff with the expertise necessary to review submitted PRAs. More detailed and broader-scope PRAs will be submitted by applicants as part of a design certification request or an actual combined operating license submittal. The staff will review these submitted PRAs.

A critical policy question is whether the staff should also generate a more detailed PRA of the prototype of each reactor. There are potentially significant benefits to be gained from such an effort. In addition to enhancing the benefits identified above for the conceptual design level PRA, a detailed staff PRA would provide a bench mark to judge the PRA submitted by an applicant. It also would provide an audit tool for evaluating the risk impact of key licensing questions. The resolution of this policy question may be affected by the degree to which the conceptual design PRA can be used to resolve licensing issues.

The risk from spent fuel accidents is currently not analyzed in detail in PRAs. The dominant contributor to risk (event frequency times offsite consequences) from spent fuel pool accidents is from seismic events. The risks from seismic events of individual early fatality risk and individual latent cancer risk are less than the Commission's Safety Goals (less than 5×10^{-7} and 2×10^{-6} per reactor-year, respectively)[5]. It is generally believed that accident scenarios involving the spent fuel would be slow acting and low frequency. The potential for drain down events is limited by the design of the pool and cooling systems. Adequate time is available to respond to loss of cooling system scenarios with alternate means of providing cooling. However, the spent fuel pool contains a source term, and the consequences of accidents, however remote, could be important if no protective actions are taken since the pool is not inside the containment. It seems

[5] NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," February 2001.

appropriate to use risk insights in the decision-making process with regard to the adequacy of the spent fuel pool in new designs.

The risk assessment of existing spent fuel pools does not appear to pose any unique challenges with regard to PRA methodology. However, it does pose some question with regard to fuel behavior under accident conditions (e.g., fuel/air interactions) and the resulting source term. There are some unique issues related to spent fuel for HTGR designs. For example, spent fuel will be kept in helium cooled tanks, and the potential for loss of helium could lead to air/water ingress accidents. However, these accidents do not appear to require any new PRA method development.

The radwaste systems do not contain sufficient radionuclides to pose a threat to the public. Thus, the risk to the public from these accidents will not be a major issue in the design evaluation. However, if risk to workers from accidents becomes a risk-informed decision criterion, then accidents involving these sources may be important.

An important decision to be made in a risk-informed regulatory environment is whether to include a worker risk metric as part of the regulatory framework. Currently, worker risks have not been included in nuclear power plant PRAs. However, they have been used at non-reactor nuclear facilities. Development of additional methods, primarily to evaluate the radiation exposure of the workers, would be necessary. This could involve more detailed assessments of radionuclide transport within a plant and more detailed human response modeling (i.e., the workers response to the accident).

The safeguard requirements at operating plants are currently based on the response to design basis threats. Risk assessments of sabotage events has not been performed due primarily to the difficulty in establishing the frequency of a spectrum of sabotage threats. This argument seems inconsistent with the fact that rare event accidents (e.g., large pipe break LOCAs) are currently modeled in PRAs even though there is no large pipe break data. Methods exist for evaluating the protection system effectiveness in preventing attackers from successfully destroying equipment to cause core damage. This equipment can be identified from a PRA. With some effort, these methods could be expanded to include the frequency of sabotage initiating events, random failure events, all failure modes of equipment (e.g., failures induced by fire, flood, and explosions), and the potential for recovery actions. Clearly, risk assessment of sabotage events can be performed if there is a desire to risk-inform the safeguard requirements. The results of PRAs can be better used to identify and prioritize equipment based on reliability and identify safe shutdown paths.

7.0 PRODUCTS AND SCHEDULE

The products (including interim products), schedule, and resources estimated for each of the tasks are identified in this section. The qualifications and areas of expertise needed to support the work are also identified. The schedule for the work allows for input from the ACRS and other stakeholders. The schedule for these activities will be integrated with the schedule requirements in the overall advanced reactor research plan.

7.1 Products

The products of the proposed PRA methods development and application tasks identified in this plan are summarized in Table 3. Specific areas of expertise needed to support each activity are also identified.

7.2 Schedule

The schedule for proposed research activities is provided in Figure 3. The current schedule for advanced reactor activities and estimates of projected available resources were considered in establishing this schedule.

Table 3. Expertise needs for PRA development and application activities.

Description of Research Activity	Products	Required Expertise
Fuel Quality and Performance	Goals and acceptable uncertainty limits	Fuel design, QA, PRA, Severe Accident Analysis, Consequence
Initiation Event Identification and Quantification	Identify initiating events and evaluate frequencies	PRA
Accident Sequence Analysis	Identify accident sequences and evaluate CDF and LERF	PRA
System Modeling		
Develop methods for modeling and quantifying the failure of passive	Method for modeling passive systems systems	PRA; thermal-hydraulic
Incorporate appropriate results from the efforts under Job Code Y6332 to develop methods for assessing the reliability of digital I&C systems	Method for modeling digital systems in PRA	PRA; I&C
Develop methods for modeling "smart" equipment in PRAs	Method for modeling smart equipment in PRA	PRA, I&C

Table 3. Expertise needs for PRA development and application activities.

Description of Research Activity	Products	Required Expertise
Data Collection and Analyses		
Identify and evaluate existing data sources for applicability to advanced reactors and determine appropriate methods to generate required data distributions	Identify and evaluate generic data sources	PRA
Generate data (component failure rates, common cause failure probabilities, and component fragilities to fire, seismic, and other harsh environments) for use in NRC PRAs	Generate data for advanced reactors	PRA
External Events	Evaluate effects of seismic events	Seismic, PRA
Accident Progression and Containment Performance		
Evaluate the current risk metrics and consider others to identify the appropriate risk metrics	Identify appropriate risk metrics	Fission Products, Consequence
Evaluate containment/confinement issues	Identify important parameters and their values for technical issue resolution	Severe Accident, Consequence
PRA Quantification		
Develop methods for addressing model and completeness uncertainty	Approach to address these uncertainties	PRA
Incorporate appropriate results from SAPHIRE to quantify full scope PRAs (Job Code Y6394)	PRA quantification code	PRA code developer
Risk Importance Measures		
Review of alternative risk importance measures/tools for ranking equipment being performed under Y6547.	Approach to identify critical SSCs	PRA
Dynamic Modeling		
Identify the capability of dynamic modeling to support the advanced reactor regulatory process	Identification of codes for use in dynamic PRA	PRA, code developers
Determine the feasibility of developing dynamic PRA tools using existing computer codes	Capability of HRA methods to support dynamic PRA	PRA; thermal-hydraulic; code developers; HRA

Table 3. Expertise needs for PRA development and application activities.

Description of Research Activity	Products	Required Expertise
Human Reliability Analysis		
Incorporate appropriate results from the development of data for use in HRA (Job Code Y6123)	Data to support HRA	HRA
Review adequacy of existing BRA methods for evaluating operator response considering multiple modules, digital I&C, and smart equipment	Assessment of HRA methods for application to advanced reactors	HRA
Safeguards and Security		
Determine the use of PRA results for use in risk-informing safeguards and security requirements	Risk data uses in safeguards and security	PRA; safeguards and security
Identify the requirements necessary to expand PRA methods to evaluate sabotage	Requirements to perform risk evaluation of sabotage	PRA; safeguards and security
Advanced Reactor PRA Development and Application		
Develop limited scope design PRA for GT-MHR	PRA for GT-MHR	PRA; reactor design
Develop limited scope design PRA for ESBWR	PRA for ESBWR	PRA; reactor design
Develop limited scope design PRA for ACR-700	PRA for ACR-700	PRA; reactor design
Develop limited scope design PRA for IRIS	PRA for IRIS	PRA; reactor design
Develop limited scope design PRA for SWR-1000	PRA for SWR- 1000	PRA; reactor design
Support development of regulations	Technology-neutral and design-specific regulations and Regulatory Guides	PRA, Licensing
Support development of risk metrics	Risk metrics	PRA
Explore use of formalized decision-making methods	Formalized decision process	PRA, licensing
Support for design certification and combined license application reviews	Resolution of design and license issues	PRA
Documentation and PRA Guidance		
Develop PRA Regulatory Guide and review guidance	Regulatory Guide and review guidance	PRA

Table 3. Expertise needs for PRA development and application activities.

Description of Research Activity	Products	Required Expertise
Develop PRA methodology document	PRA methodology document	PRA

Figure 3. Advanced Reactor PRA Research Plan Schedule