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April 12, 2001

Dr. Ashok Thadani
U.S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

Dear Ashok:

As requested by you at the 1999 WRSM, the national laboratories that have traditionally provided significant support to the NRC Office of Research have created a set of recommendations for future nuclear power plant safety research. We are all pleased to present the final version of ***Future Directions for NRC Research DOE National Laboratory Recommendations***, dated April 2001.

This material was developed largely through phone and emails communications among the labs, which precluded any significant degree of merging or rank ordering of ideas. We did, however, create a general discussion for nine themes that appear throughout the individual laboratory contributions. Several documents are now available providing research advice from various viewpoints -- ACRS Report on Reactor Safety Research, Rogers' Committee on Role and Direction of Nuclear Regulatory Research, NERAC Subcommittee for Long Term Planning for Nuclear Energy Research, Joint DOE-EPRI Strategic Research and Development Plan to Optimize U.S. Nuclear Power Plants (NEPO), and now the national laboratories Future Direction for NRC Research. We suggest that NRC/RES consider as a next step the integration of these source materials into a larger coherent picture, utilizing risk perspectives to help prioritize. The laboratories would appreciate being included in such a process.

Please feel free to contact me or communicate directly with the points of contact listed in each laboratory's contribution for further information or discussion.

Sincerely,
/original signed Tom/

Future Direction for NRC Research
DOE National Laboratory Recommendations
April 2001

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1. Introduction

At the invitation of Dr. Ashok Thadani, NRC Research Office Director, a number of DOE national laboratories having long-term relationships with the NRC have created a set of recommendations for future research. Participating laboratories include:

Argonne National Laboratory (ANL)
Brookhaven National Laboratory (BNL)
Idaho National Engineering and Environmental Laboratory (INEEL)
Lawrence Livermore National Laboratory (LLNL)
Los Alamos National Laboratory (LANL)
Oak Ridge National Laboratory (ORNL)
Pacific Northwest National Laboratory (PNNL)
Sandia National Laboratories (SNL)

Each laboratory contributed several pages describing their individual views on future research that could help the NRC continue to successfully execute its nuclear regulatory mission. These are organized by laboratory in alphabetical order in Section 5. While no attempt has been made to merge, peer review, check completeness, or prioritize these recommendations into one list, at the urging of Dr. Thadani, an attempt to summarize the suggested research around common threads is presented in Section 4. Prior to this summary, a short discussion on approaches to designing and conducting future research is presented as Section 3.

It is the intent of the Laboratories that the collective technical experience and insights presented here will be a significant resource for the NRC Office of Research as they provide leadership of future nuclear regulatory research. The Laboratories are strongly committed to continued support of NRC research and welcome opportunities to discuss the contents of this report.

2. Overview

The nuclear industry in the United States finds itself at a crossroads. Existing nuclear power plants (NPPs) require technologies to remain safe and to maintain public confidence as they age, while the next generation of plants must implement technologies to gain public confidence for support of renewed construction. In the restructured electric power industry, new technologies to decrease the construction and operational costs of nuclear power plants will be needed to make them more economically competitive. Many believe that regulations and the regulatory process must change to support the development of the next generation of nuclear power, while ensuring that the existing nuclear power industry and infrastructure becomes even more efficient, safe, and reliable.

In the context of this legacy of current nuclear power technology and regulations plus an uncertain future of yet-to-be-defined advanced nuclear power technology, there is a need for a broad-based research effort to provide the technical underpinnings of both prescriptive and risk-informed regulations. This needed regulatory research can be thought of as falling into three categories.

One category would address pressing regulatory questions associated with the continued safe operation of current nuclear facilities. This would include topics ranging from plant aging, condition monitoring, and advanced I&C systems, to radioactive waste and nuclear fuel storage and disposal.

A second category would focus on the need to improve regulations through an evolutionary risk-informed regulatory process for existing nuclear power technology. These improved regulations would allow nuclear facilities to be operated and maintained more efficiently by taking advantage of our more complete understanding of safety margins and accident scenarios.

Finally, a third category would extend the risk-informed regulatory process and technical insights to a new set of regulations and research program for perhaps radically different next-generation power systems. Although the exact form of the next-generation of regulations remains ill defined, it will likely include performance and risk-based standards for advanced power generation and waste disposal systems. These advanced systems may include modularized reactor systems, accelerator driven systems, new proliferation-resistant fuel cycles, waste transmutation and recycling technologies, streamlined design-to-construction information management systems, and "inherently" safe reactor concepts. To adequately prepare for the licensing of these future reactors, the NRC must conduct research to ensure that its staff have sufficient technical insights to support a licensing process that will maintain safety and public confidence.

These three categories served as general criteria for creation of the individual laboratory recommendations contained in this report, although the write-ups do not adhere to a single strict format and many research subjects span more than one category. These categories did form a useful basis for grouping recommendations into themes, which enabled the summaries in Section 4.

3. Approaches to Designing and Conducting Research

The NRC has a successful history of conducting research that has developed over more than two decades. The exceptional safety record of the nuclear industry is in no small part the result of integrating research into the licensing and oversight of nuclear power plants. In particular, the research begun after TMI-2 provided the technical basis for much of our current understanding of improved operational performance and the ability of the plant and its operators to respond to off-normal events and conditions. Advances in our understanding of thermal-hydraulics, reactor physics, materials behavior, structural performance and response, plant aging, human performance, reliability, risk assessment, and the capabilities and vulnerabilities of the plants under postulated severe accident conditions are several of the key accomplishments in this history of safety of reactors and their byproducts. This work has allowed NRC and the industry to examine the current regulatory fabric in order to determine whether 1) safety can be improved, 2) the regulatory process can be made more effective and efficient, 3) there is undue burden on the licensees, and 4) communication with stakeholder groups can be improved. One very important tool that has been used to focus on these topics is probabilistic risk assessment.

Over the past few years there has been a significant effort to risk-inform the regulations. With the move towards risk-informed regulations, the NRC is establishing itself as a leader among federal agencies in focusing not only its own resources but also the resources of the entities that it regulates on issues that are the most important to the public – in this case safety. Nothing short of the existence of the nuclear enterprise in the U.S. and our continued role as the world leader for nuclear technology is at stake. We cannot afford the loss of a technology that presently provides about 20% of the electrical production in the U.S. and which may be a key contributor to energy independence without emissions of greenhouse gases and other pollutants to the atmosphere. Therefore, we encourage the NRC to continue, and to expand, research that will anticipate new reactor technologies and will also assure that the current fleet of light water reactors remain reliable and safe, especially since many of them will be relicensed. The use of risk information to aid in prioritizing future research funding is strongly encouraged by the Laboratories. At the same time, we encourage the increased involvement of all other stakeholders in the nuclear enterprise. Valuable input and participation by others will help to promote understanding of the risks and benefits of nuclear technology.

<u>Existing Nuclear Power Plants</u>

Current risk information has been very valuable in guiding recent regulatory changes. However, it is recognized that these risk assessments are incomplete in some areas and very uncertain in others. Research to improve this situation will allow further and more effective regulatory changes. The experience of individuals in the NRC, the Laboratories, academia, other contractors, and the commercial nuclear industry can be used to identify the major shortcomings and prioritize the areas where successful research is likely. This is particularly true for existing NPPs, where a wealth of knowledge had grown over the last fifty years. Therefore, we encourage the NRC to turn to this broad range of experts in defining research on existing plants (the request for this document is an indication of the NRC's desire to do so). One approach is workshops that pull together representatives from each sector. For example, the Department of Energy's Office of Nuclear Energy, Science and Technology (DOE-NE) has an annual meeting to help define and prioritize the research to be conducted as a part of the program for Nuclear Energy Plant Optimization (NEPO). Several of us have found the process to be a good one. The NRC participates but should define its own process for seeking regular (annual, as a minimum) input on research aimed at existing NPPs. Perhaps this could be associated with some regular meeting that desired participants typically attend, such as the WRSM.

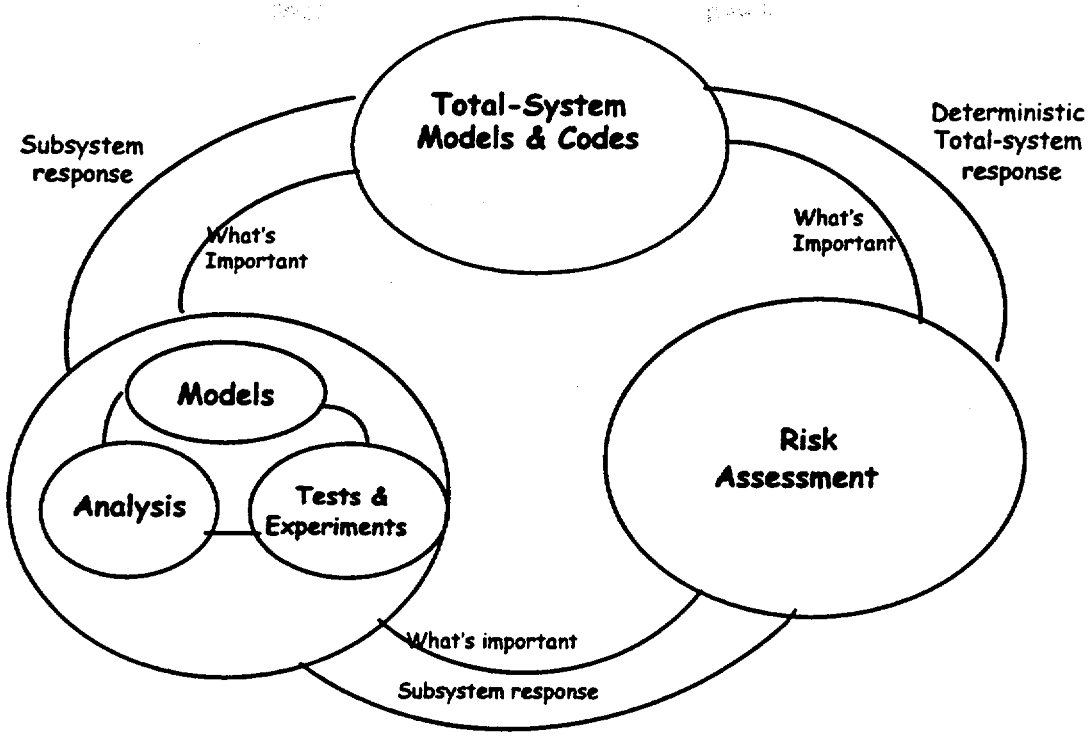
Improved Regulations

As noted above, the continued movement to risk-informed regulations is essential to the future of the nuclear industry and the NRC. Several of the contributions that follow include specific improvements to the regulatory framework and we encourage their consideration. We believe that changes, particularly proposed relaxations, in the current regulations will continue to raise questions that can only be answered by a strong research program with the capability to look at the integrated effects of changes on NPP safety. Progress on risk-informing regulations can serve as a focus for future deterministic research in areas important to safety.

Advanced Systems

The current regulatory process has evolved over 40+ years beginning with the development of defense-in-depth concepts in the 1950s and including landmark regulations such as the siting criteria (10CFR100), the general design criteria (10CFR50, Appendix A), and the ECCS rule (10CFR50.46). For the most part, these regulations were developed without the benefit of extensive severe accident and risk assessment research input. As a result, the regulations were developed in a necessarily conservative fashion. After TMI-2 major research activities were carried out that are now allowing major regulatory changes for current reactors. It is not necessary to repeat the 40-year history for new plants. We now understand how to develop a systematic framework for regulation (a "clean sheet of paper" approach is encouraged) that is derived from safety goals and risk information, but depends on detailed modeling and experiments to assure the necessary margins of safety.

The following figure depicts the manner in which different research elements can interact. A risk framework can provide a structure for identifying information needs to complete a risk assessment. A sub-cycle of modeling, analysis, and experiments helps define an understanding of subsystems that may be important to safety. This understanding of subsystem performance is integrated into total system models and codes that can be used to predict the deterministic behavior of the entire system for a given set of conditions. Both the subsystem modeling-analysis-experiment loop and the total system analyses codes are used as input to probabilistic models and analyses that quantify the risks associated with the system. The results of the risk analyses in turn help define areas requiring improved understanding in both the subsystem and total-system models, i.e., they help define future research.



Comprehensive Systems Research Approach

Where do we begin in this cycle? For a new reactor type, the first steps are to establish risk targets and safety themes to guide an appropriate regulatory process. In order to validate such a process, a research program must be developed that incorporates the insights from experts in all of the areas represented in the above figure. For example, safety goals can be defined appropriate for a given plant type. Then the risk contributions can be allocated to the major plant functions, such as reactivity control, heat removal, containment, etc. Research would go forward to ensure that plant performance is likely to meet the desired goals in each area or to reallocate risks appropriately. Experts can begin the process by providing insights into plant behavior from expert judgment and best available technical information. Identification of key uncertainties should be a focus of initial activities. This process necessarily lends itself to a team approach, where the best experts available are pulled together into a team that provides information within the context of the new regulatory framework and plant performance goals.

It is clear that research activities should include model development, experiments, tests, and analyses to gain insight into subsystem and phenomenological behavior. It will also include the modification of existing total-system codes and/or the development of new codes for answering questions of deterministic total-system response to specific conditions. Although a licensee may have primary responsibility for performing probabilistic risk assessments, the NRC should assist in developing more complete treatments of risk and should perform assessments of their own to confirm industry results and clarify the critical issues for licensing.

We are concerned that recent trends in our laboratories have been to reduce or eliminate tests and experiments because of funding constraints. While we understand that large scale testing will be limited, a certain amount of model validation and phenomenological investigation will always be required. Therefore, we recommend the following. First, use a framework-based approach as discussed above to identify areas where key uncertainties exist in plant and system behavior. In many cases, some fundamental experiments will be necessary to develop constitutive models and enhance the understanding of subsystem behavior. Second, the NRC must take advantage of improvements in analytical capabilities that have occurred in our organizations. While this will require large investments in modifying and applying these tools to advanced nuclear energy systems, the investments are likely to be much smaller than required for numerous large-scale experiments. Finally, validation tests and experiments are critical to the overall credibility of the models, and hence, the risk assessment results. Maintaining appropriate facilities must, therefore, be a vital part of our research program.

4. Recommendation Summary: Research Themes

All the laboratory ideas were binned by subject and primary applicability -- current reactors, regulations, or advanced reactors. Some nine themes emerged, and a high level summary was prepared for each. Order of themes in this section implies no priority. Further specificity of research details or priority would require some sort of group process involving appropriate experts, possibly including a workshop, utilizing the risk based approaches discussed in Section 3.

<u>Computer Code Improvements and Expansions</u>

The desired end result of much of the research described in this document is the incorporation of the improved understanding of a system's behavior into an analytical tool, often including a computer code, that can be applied to existing or future NPPs. Sometimes, a computer code that models systems or complicated phenomena is so complex that the code development itself becomes a research effort. One such area is the continuing development of thermal-hydraulic (T-H) codes. The use of existing codes typically involves conservative assumptions and inputs, which results in a conservative answer, but the degree of conservatism or uncertainty cannot be determined. Therefore, for the T-H codes, approaches that propagate initial uncertainties through the calculations are needed. Also needed is further integration with fuel material performance and neutronic models. For fuel neutronic behavior, the issue is one of level of detail. A fuel assembly code that calculates the details of neutronic behavior on a pin-by-pin basis is needed to provide input to existing and future transient codes.

With the development of new reactor concepts comes the need for codes that can model behaviors for new systems for which a phenomenological understanding exists. For example, modeling the 3-dimensional flow through a pebble-bed reactor is necessary to define the temperatures of individual fuel elements. Such a calculation may be performed with existing codes, but accident conditions may require capabilities that don't presently exist. Also, because pebble-bed and other reactors may rely on the fuel coatings for the containment of fission products, the ability to 3-dimensionally model the coupled T-H-neutronic behavior of elements with inhomogeneities may be required to limit testing to a reasonable level.

With the extraordinary development of computer systems, we have an opportunity for new reactor concepts that is unprecedented – the virtual reactor. The ability now exists to put together total system models for entire reactors that include the behavior (thermal, neutronic, structural, electrical, etc.) of each subsystem, including fuel, coolant, shielding, energy conversion, and instrumentation and control systems. Even human factors could be modeled and integrated into the total-system virtual reactor. With such models, parametric studies would provide invaluable insights into tradeoffs. One necessary offshoot of such models would be severe accident codes for the future reactors, which might not require all the systems of the virtual reactor, but would require expanded phenomenological and parametric development for specific subsystems.

Fuel Behavior

Suggestions for research on fuel behavior generally deal with extended use of existing or modified fuels (fuel burnup), new fuels or new storage situations. Among the issues for high burnup are the use of advanced cladding alloys (accident conditions), the characterization and modeling of the rim restructuring phenomena (storage and accident conditions), high burnup fuel foaming (severe-accident conditions), and high-burnup-fission-product releases (severe-accident conditions). With respect to new fuels, research is needed for the behavior of mixed oxide fuel (MOX) under accident and storage conditions and the performance of coated fuel for pebble bed modular reactors, including questions of containment, porosity for cooling, and the effects of inhomogeneities. Storage issues include verifying the integrity of cladding in dry storage and testing to verify behavior in a repository.

Aging and Integrity of Structures, Systems, and Components

With license renewals and plant life extensions, issues of aging structures, systems, and components (SSCs) will require emphasis. The potential for cracking and resulting loss of integrity is of particular interest for aging pressure-bearing components. Continuing work in the areas of environmentally assisted cracking due to radiation is suggested, as is work in corrosion and stress corrosion cracking, which has limited light-water-reactor performance for thirty years. In related work, void swelling in PWRs may lead to embrittlement and should be investigated. The completion of current rules for ensuring the integrity of reactor pressure vessels (RPVs) under pressurized thermal shock (PTS) scenarios is a high priority. Although PTS concerns have led to the collection of data on flaws in welds in RPVs, research is still needed for similar information in piping and spent fuel casks. A more general problem is the state of probabilistic fracture mechanics, where developments in both methods and data on crack-growth rates have not been incorporated into codes being used, and where several follow-on research activities are possible.

Another critical area is radiation embrittlement of steel components, including RPVs. Research needs to continue on phenomenological and mechanistic scales to understand and predict embrittlement effects due to long-term neutron exposure. This includes the study of through-vessel wall attenuation, effects of high fluence level (consistent with license renewal periods), synergism between thermal aging and neutron embrittlement, and embrittlement after annealing. Assessing the embrittlement characteristics of the proposed Generation IV structural materials is also critical.

Aging is an important issue with other SSCs, including structures, passive components, electrical systems, and valves. Unlike active components whose integrity can be monitored through their performance characteristics, degradation of structures and passive components may not be easy to detect and these components have not received as much attention in the past as active components. Aging can reduce the available margins and can cause changes in engineering properties, structural capacity, failure modes, and locations of failure initiation. Although cable aging has been studied for years, fuses, connectors, terminations and electrical penetration assemblies require in-depth evaluations of the practices used for their qualification and continued use during license renewal periods. Among the many unanswered issues with respect to the aging of valves are the effects of preconditioning for extrapolating to design-basis conditions, the ability of diagnostic tools for motor control centers to monitor degradations, gearbox efficiency, the accuracy and adequacy of air-operated diagnostic tools, the effects of corrosion and friction on certain components, and stem lubrication with aging.

Perhaps the most effective way to deal with many aging issues is through improved monitoring, surveillance, and diagnostics. Recent PWR problems suggest that in-service inspection for detecting and characterizing degradation can be further improved. By assessing commercially available systems and foreign experience, the surveillance and diagnostics approaches for key reactor and steam-supply-system components may be improved. Work on the imaging science of welded components might lead to significantly improved NDE inspection results. Progress in instrumentation and sensors makes the real-time monitoring of safety-critical systems, like the containment, possible. This creates the possibility of an "open" network of information to the NRC and the public in the event of a natural disaster like an earthquake.

Risk Assessment

The development of a complete set of probabilistic risk assessment (PRA) standards is a high priority. This development requires research in applying PRA in a number of areas, including digital instrumentation and control (I&C) (including the emerging area of cyber security), human reliability, plant aging, fire risk, effects of aging, and low power/shutdown. Although these areas may be research areas in their own right, particular attention to their application in PRAs is needed. For example, I&C details are typically not included in PRA evaluations. An assessment tool to automate failure mode and effects analyses is needed. Also, data from research on containment integrity can provide improved representation of containment failure and it's associated uncertainty in PRAs.

Additional work on uncertainty in risk assessments will aid in defining the ultimate use of PRA in a robust decision making process. An area that has not received much attention is latent failures, including the effects that maintenance, surveillance, and work practices have on such potential failures. The specialized area of seismic PRA will require research to make it useful in future risk-informed applications.

It has been more than a decade since NUREG-1150 was developed and a rebaselining of risk is warranted. In addition to advances in methodologies and the availability of operational data, there have been changes in regulatory oversight, maintenance practices, and operations. The move to risk-informed regulations and PRA standards underscores the need for rebaselining. One area that has received little attention is the historical consequences of radiological accidents and events; pulling this data together for a rebaselining is critical. A rebaselining would improve decision-making, provide focus for research, provide comparisons for industry efforts, and enhance communication with the public. In particular, for the last item, the NRC should commission a study to examine discounting the risk due to radionuclides with short half-lives with respect to the timeframe of interest as opposed to chemicals with infinite half-lives.

Human Factors

As discussed in the previous section, the role of human factors in PRAs requires more research; but with potential changes in the control room, the role of humans is changing and will require evaluation. In particular, the move to digital I&C may reduce the potential for certain human errors but may also introduce other, previously unidentified errors. Similarly, the potential impact of latent human errors (maintenance, surveillance, work practices, etc.) needs to be assessed with further research before it can be included properly in PRAs.

With the increased insight into human factors that has occurred since the last NPP was constructed in the U.S., it is reasonable to hope and expect that the next generation of NPPs will include human engineering in all facets of operation. For example, the control room of the future will utilize digital I&C (i.e., computers and software) in ways that have not been previously seen. The large potential benefits for improved efficiency, monitoring effectiveness, and control will bring with them the potential for failures, including human failures that are not possible in existing plants.

Regulatory Framework

Although the NRC is moving aggressively to incorporate risk insights into the regulatory process and past research has been highly successful, much remains to be done before the full benefits of risk-informed regulations along with our significantly improved understanding of systems and materials can be achieved. In addition to a large potential payoff with safety benefits, the risk-informed process can lead the way for reducing unnecessary conservatism and significantly lowering costs for both the NRC and the industry. Some of the areas for which research is required for improving PRAs will, in turn, need to be incorporated into the risk-informed process. These include digital I&C and fire risk. The NRC and industry have been unable to reach agreement on the adoption of a standard on performance-based fire protection for NPPs and developing a framework for implementing risk-informed fire protection should remain a priority. Also, a concise guide on methods of fire-risk analysis is needed.

An area of potential regulatory improvement that has not been significantly addressed is radio-bioassay and internal dosimetry. It is recommended that the NRC assess the need and impact of a National Voluntary Laboratory Accreditation Program for personal air sampling and for radio-bioassay measurements, when those results are used to compute the dose of record. Also, a benchmarking performance assessment of Bayesian methods for bioassay and internal dosimetry is needed to reach consensus on their use in areas regulated by the NRC. To reduce potential over-conservatism, the NRC should also perform research to determine the impact of uncertain dose estimates in regulations.

There have been many NRC research and regulatory programs conducted over the last twenty years to study the capability of nuclear facilities to withstand the effects of natural phenomena. What is now needed is a comprehensive and integrated review of *all* existing regulatory guidance related to the effects of natural phenomena. The knowledge gained by many experienced NRC staff and contractors needs to be incorporated in a systematic way to assure regulatory effectiveness, efficiency, and realism, and to reduce unnecessary licensee burden for the future.

A new framework for advanced non-LWR reactors is a major challenge for the NRC. Although there are differences of opinion as to the details of an approach, a common thread is building in risk concepts at the start. The current regulatory framework for General Design Criteria and design certification is inadequate and a complete revision of relevant requirements is needed. A top-down generic framework that builds off the current Safety Goal Policy and that can be adapted to each new design should be developed. One suggestion would include focusing on the seven "cornerstones" of safety of plant operations: initiating events; mitigating systems; barrier integrity; emergency preparedness; public radiation safety; occupational radiation safety; and physical protection. Another suggestion is that the regulatory framework includes General Safety Requirements, a Framework and Content Guide (for the specific reactor type), and related detail review plans. A proven tool for supporting the allocation of resources for preparing for license reviews is the Phenomena Identification and Ranking Tables (PIRT) process, which can assist in the assessment of the adequacy of computational tools and data.

Safety and Cyber Security of Digital Instrumentation and Controls

The computer and Internet revolution has largely bypassed nuclear power plants. But, as the nuclear industry incorporates modern digital I&C and computer technology into existing and future plants, the industry faces ever-increasing vulnerability to digital system reliability/performance and cyber attack/intrusion. The obsolescence of existing analog components increases the likelihood of the use of digital technology in safety-related I&C applications. Potential areas for research into safety-related applications include: reliability of both hardware and software; software qualification; software diagnostics; communications architectures, especially wireless networks; in-service monitoring and prognostics; and environmental effects. The effects of fire on digital I&C is an area of particular concern.

With the inclusion of computer-based systems, cyber security from either insider or outsider attacks becomes a real issue. A need exists to identify and assess the vulnerabilities of computer-based systems whose reaction, failure, or malfunction due to insider or outsider attacks would prevent the performance of the safety function, adversely affect physical security measures, or degrade, impair or prevent operations. The Department of Energy is commencing a small program to address this issue, but the NRC has only established an interface with the program. Participating in funding this program would increase the potential that rulemaking on cyber security will be predicated on a sound foundation.

Repositories/Storage

Dry storage is becoming increasingly common and the safety of that storage may require new research. It has been suggested that the present level of drying may not always be adequate and mechanisms that degrade the cladding and fuel may be at work. Also, there is a need to verify the integrity of cladding of fuel in storage. With respect to the Yucca Mountain Project (YMP), the NRC may need to verify models, including models of fuel/cladding/waste package degradation based on experimental results, that the DOE is proposing. Differences between irradiated materials and tested unirradiated materials may be particularly important. Similarly, changes in cladding performance and fuel isotopic content as fuel burnup increases need to be understood. Cooperative research and data sharing between NRC, DOE, and EPRI should continue. The long-term performance of nuclear waste containers will likely be an issue during the licensing of a repository and the NRC needs an underlying research basis for accepting/rejecting the DOE's models of performance.

Other Phenomenon or System Research

A number of identified research areas are unique to Boiling Water Reactors (BWRs). They include: the impact on potential severe accidents of higher decay heats associated with power upgrades; the behavior of partial-length fuel; the consequences if a BWR core is uncovered; the potential failure of Mark I liners; high-pressure melt injection; the end-of-cycle flux shape; and feedwater flow venturi calibration (also a PWR concern). Other LWR accident issues include: scenarios involving fuel rods exposed to air, including the interaction of air with residual fuel following reactor vessel rupture (relates to PHEBUS-FP experiments); the deposition of aerosol particles on surfaces within the secondary side of steam generators; the effects of electrostatic charging on the behavior of nuclear aerosols; and the coolability of molten core material which has penetrated the reactor pressure vessel and is undergoing Molten Core Concrete Interaction (MCCI) with the containment basemat.

5. Individual Laboratory Contributions

Individual laboratory recommendations for future NRC research are arranged in alphabetical order in this section. As noted earlier, no attempt has been made by the Laboratories to merge, peer review, check completeness, or prioritize these ideas. Recommendations are based upon research experience and expert judgement at the individual laboratories. A point of contact is listed at the beginning of each input.

Research Relevant to NRC Licensing Issues Argonne National Laboratory

April 11, 2001

Contact: Michael Billone, (630) 252-7146, billone@anl.gov

Introduction

Argonne National Laboratory (ANL) has several on-going NRC-sponsored research programs designed to address important licensing issues. With respect to fuels and materials compatibility, current research includes: 1) adequacy of current LOCA and RIA acceptance criteria for high burnup fuel with traditional Zircaloy claddings, 2) criteria for license renewal for medium burnup (≤45 GWd/MTU) fuel stored in dry casks, 3) criteria for high burnup (>45 GWd/MTU) fuel stored in dry casks, 4) criteria to ensure integrity of reactor components subjected to environmentally assisted cracking (EAC), and 5) criteria to ensure integrity of steam generator tubing subjected to a variety of degradation mechanisms. With respect to severe accidents, NRC has sponsored research related to 1) material properties, 2) in vessel melt progression and coolability, 3) ex-vessel MCCI thermal-hydraulics, fission product release, and debris coolability. In addition to the current scope of these programs, there are logical extensions of these programs into areas that involve newer materials, as well as extensions of the programs to include issues associated with Next-Generation Nuclear Reactors. Both continuation and extensions of the current programs are summarized below. These research areas are intended to enhance NRC's ability to make risk-informed decisions regarding licensing issues for existing as well as advanced reactor designs and related technologies.

Confirmatory Research

1. High burnup issues for advanced cladding alloys

Advanced (relative to Zircaloy) cladding alloys include Zr-1Sn-1Nb (e.g., ZIRLO) and Zr-1Nb (e.g., M5). Although these alloys appear to be superior to Zircaloy performance in the area of reduced in-reactor coolant-side corrosion and associated hydrogen pickup, it has not yet been demonstrated that the current NRC LOCA criteria (e.g., Equivalent Cladding Reacted [ECR] ≤ 17% and peak T ≤ 2200°F [1204°C]) are adequate to ensure integrity during the ECCS quench and to ensure residual post-quench ductility. The same testing techniques that are used for Zircaloy can be used for the advanced claddings to directly test the adequacy of the current criteria, as well as to improve on the criteria if necessary. Such tests would include high temperature steam oxidation kinetics studies, LOCA criteria testing, and determination of mechanical properties. These mechanical properties would be tested at moderate-to-high strain rates so that they would be applicable to the analyses of both LOCA and RIA events.

2. Impact of radiation-induced phenomena on reactor component crack growth rates

This is a continuation of the current work. However, the effects of changing water chemistry, the effects of improved alloy compositions and thermomechanical treatment, the effects of industry-proposed mitigating measures, and the resulting inspection schedule need to be continually evaluated in terms of crack growth rates in a radiation environment. EAC is an on-going problem that needs more research to address the issues.

3. Steam generator tube integrity

Similar to item 2, this is an on-going problem involving detection and characterization of degraded tubes, evaluation of structural integrity and leakage rates of degraded tubes, and degradation rate within an inspection cycle.

4. Severe Accident Research

In the wake of the TMI accident, the NRC initiated significant experiment programs to investigate phenomenology associated with core melt accident progression in LWR systems. Information gathered from these experiments has formed the foundation for the development of computer codes for the calculation of accident progression in nuclear power plants. Although extensive progress has been made in this area, a few technical issues have not been fully resolved and warrant further consideration. One such issue concerns the coolability of molten core material, which has relocated from the RPV to the reactor cavity region and begun to interact with underlying structural concrete. If unchecked, this sequence could eventually lead to containment failure either by basemat penetration or by overpressurization. Thus, SAMGs for operating LWRs include the potential for flooding the reactor cavity in the event that the core melt penetrates the RPV. In the past, NRC has been a leader in research investigating the efficacy of water in quenching and thermally stabilizing an ex-vessel core melt accident. Currently, NRC is leading efforts to institute an OECD-sponsored research project at ANL to resolve the ex-vessel debris coolability issue. Achievement of this objective would provide the technical basis for evaluating SAMG's for existing plants, and would also aid in the development of better containment designs for advanced plants. To fully realize the benefits of this research, model development needs to be conducted in parallel with the experiments in order to provide validated computational tools for extrapolation to plant conditions. This is deemed to be a critical aspect of the research, since much of the phenomenology involved with debris coolability is new and is not modeled in any of the system-level codes.

Anticipatory Research

Recent energy shortages in California have highlighted the need for advanced, non-polluting energy supply systems. This need has created the potential for a revival in the nuclear power industry since this energy source is capable of providing cost-effective electricity with zero air emissions. However, there are lingering concerns in the general public regarding nuclear power, particularly in terms of plant safety and radioactive waste. To address these and other issues, there are substantial worldwide efforts currently underway to develop reactor designs which are economical, inherently safe, simple to operate, and proliferation-resistant. A second significant area of research related to new reactor technology is the development of accelerator-driving subcritical reactor systems for waste transmutation. The national Laboratories and industry will continue to support the development of these advanced reactor concepts. In fact, conceptual designs for many of these technologies are sufficiently mature that safety-related licensing issues are being defined. For instance, one class of new reactor designs features the use of heavy liquid metal coolant to achieve significant improvements in plant economy, operational simplicity, inherent safety, and proliferation resistance. Moreover, a candidate target design for the ATW test stand also employs heavy liquid metal as the target material. Thus, both of these systems share many of the same concerns regarding materials compatibility (corrosion). New fuel designs are also under consideration for STAR-LM. Proactive safety research in these areas will provide the technical basis for effective licensing of these and other new plant designs. Other research areas which would better position NRC to carry out risk informed licensing are highlighted below.

1. Near term research on impact of MOX fuel

ANL has extensive technical expertise in the area of MOX fuel. In addition to characterizing the properties and behavior of MOX fuel, particularly during an RIA, ANL is interested in MOX fuel cladding behavior during LOCA and dry cask storage.

2. Pebble Bed Modular Reactor (PBMR)

ANL has interest in issues associated with the Pebble Bed Modular Reactors. Nuclear materials experts at ANL have been briefed on the concept and are in the process of defining ways in which they could contribute to resolving some of these issues that would affect the establishment of regulatory criteria for such reactors.

Issues Requiring NRC Future Research Brookhaven National Laboratory

February 28, 2001

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This white paper identifies issues that Brookhaven National Laboratory believes are important for NRC/RES to address in the near future. Most issues are of interest independent of the reactor type being considered and are, therefore, relevant to both current nuclear facilities and those that might be proposed for the future. However, at the end of the paper are issues related directly to the PBMR. No consideration was given to prioritizing the list or assuring that it was all encompassing. We recognize the need for using risk measures as an aide to prioritizing research projects.

I. GENERAL ISSUES

1. Calculational Capability for Fuel Neutronic Behavior

A calculational capability is needed in order to be able to calculate fuel assembly properties. NRC/RES has supported the development of the PARCS code, which calculates the neutron power/flux throughout a reactor core. However, to calculate the details within an assembly one needs a fuel assembly code (a.k.a. lattice physics capability). This type of code calculates the details of the power on a pin-by-pin basis and provides the nuclear data, which is input to transient codes such as PARCS. Currently the only way for NRC to have a core model for PARCS is to obtain it from some outside source. Having a calculational capability would enable NRC to generate its own data for PARCS, including pin peaking factors, and would enable NRC to work on safety problems with new fuel types that may be of interest in the future (e.g., MOX fuel, fuel with thorium, fuel with new materials, and new assembly designs).

2. Human Factors

Human factors research is needed in the areas of control room technology, human performance, and risk impact of advanced technology. The industry will see significant developments over the next decade that will have important implications for the staff's capability to review control room designs. First, with the license extension program well underway, the U.S. reactor fleet is looking at modernization programs to ensure their productive and reliable operation for another 20 years. A centerpiece of modernization will be digital instrumentation and control (I&C) system upgrades that include extensive control room modernization. The shift to digital systems and computer-based control room technology will have a very significant impact on plant operations, changes in personnel roles and responsibilities, and potentially plant safety. The second development is the interest in industry on more advanced reactor technology. The DOE is funding research into "Generation IV" concepts and industry has asked the NRC to review the PBMR. Control room and operational considerations will be very different for these designs relative to the current and advanced LWRs plants with which the staff is familiar. While the NRC has developed guidance for the human factors (HF) aspects of control room reviews, the issues associated with these two industry trends will require work to focus on the unique HF aspects of large modernization programs and alternative reactor technology designs. The NRC will also need to develop methods to better address the effects of digital systems and computer-based control on human error and risk assessment

3. Aging Degradation of Structures and Passive Components

The past performance of structures and passive components at nuclear power plants (NPPs) has been good. However, as these structures age, incidences of degradation due to various aging mechanisms are likely to increase the potential threat to their functionality and durability. Unlike active components whose integrity can be monitored through their performance characteristics, degradation of structures and passive components may not be easy to detect and these components have not received as much attention in the past as active components. Aging can reduce the available margins and can cause changes in engineering properties, structural capacity, failure modes, and locations of failure initiation. Therefore, an understanding of how aging affects the structural performance of degraded structures and passive components is essential to ensure the continued safe operation of NPPs. Analytical methods must be enhanced, and/or developed, where needed to create the tools necessary to predict the behavior of aged components under various loadings. These analytical methods can then be utilized to perform deterministic and probabilistic analyses to determine what levels of degradation will not significantly affect the safety and intended function of the structural components. Such research has been performed for degraded reinforced concrete members and is currently in progress for buried piping. It is recommended that this research be continued for additional components such as anchorages, tanks, masonry walls, and other structural components, which have been prioritized already in a past scoping study. The results of such studies would provide the technical basis for acceptable analytical methods, acceptance criteria, and guidance documents for addressing issues related to aging degradation of structures and passive components. These results can greatly assist the NRC staff in various licensing activities such as license renewal, plant maintenance reviews, evaluation of Licensee Event Reports, plant audits and inspections, and risk-informed regulatory decisions.

4. Aging Research on Electrical Equipment

With the recent interest in extending the license term of NPPs, identifying and managing the effects of aging during the period of extended operation is becoming increasingly important. A good deal of research has already been performed to provide an understanding and technical basis for regulatory decisions related to the continued operation of components and systems performing safety functions, however, a number of issues still remain to be resolved. One category of components that requires additional research is electrical equipment. These are typically passive components that receive little attention in terms of maintenance and monitoring.

Electrical cables have been the subject of research for the past several years, and a great deal of information has been obtained related to the effects of aging on their performance. Of particular concern to both regulators and utilities is whether there are effective condition monitoring techniques available that can be used to determine the current condition of cables in situ, and predict their future accident survivability. While advances have been made in identifying promising condition monitoring techniques, additional research is needed to demonstrate how these techniques can be used to successfully predict future performance, and what requirements might be imposed for license renewal. This research would be useful in regulatory decision making regarding requirements to implement some form of condition monitoring during periods of extended operation.

In addition to cables, there are other electrical components that perform important safety functions and are susceptible to aging degradation, such as safety-related fuses, electrical connectors and terminations, and electrical penetration assemblies. While preliminary aging studies have been performed on some of these components, in-depth evaluations of the practices used to qualify these components for use in harsh environments have not been performed. Also, methods of monitoring their condition in situ and predicting their accident performance have not been evaluated. Additional research is warranted on these components to determine if regulatory concerns exist regarding their acceptability for continued performance during license renewal periods.

5. Seismic Probabilistic Risk Assessment Methodology

Seismic probabilistic risk assessment (SPRA) methodology has been extensively used by the nuclear industry in performing the NRC Individual Plant Examination of External Events (IPEEE) program. After detailed reviews of the licensees' submittals summarizing the SPRA results, the following research areas have been identified in order to make the SPRA methodology a useful tool for future risk informed applications:

- Guidance needs to be developed on the selection of the anchorage for the Uniform Hazard Spectrum (UHS).
- The appropriateness of the industry screening practice, which is based on the use of the review level earthquake and the UHS shape, should be closely examined.
- There is a need to develop guidance as to the acceptable approaches that may be used to perform simplified fragility analysis.
- There is a wealth of information that is now available to develop a comprehensive database on the fragility of nuclear plant structures, systems and components.
- Further work is needed to better define the relationship between the plant HCLPF (high confidence of low probability of failure) capacity developed using the SPRA methodology and the licensing basis capacity, which is based on deterministic methods.

6. Improvement of Regulatory Guidance Related to the Effects of Natural Phenomena

There have been many NRC research and regulatory programs conducted over the last twenty years to study various aspects of the capability of nuclear facilities to withstand the effects of natural phenomena. Extensive knowledge has also been gained by studying the effects of major events (hurricanes, tornadoes and earthquakes) on industrial facilities. In addition, data from large-scale testing has been obtained, primarily from Japan. Selected information has been used to update various aspects of existing regulatory guidance. However, what is now needed is a comprehensive and integrated review of all existing regulatory guidance related to the effects of natural phenomena. The knowledge gained by many experienced NRC staff and contractors needs to be incorporated in a systematic way to assure regulatory effectiveness, efficiency and realism and to reduce unnecessary licensee burden for the future.

7. Rebaselining of Risk for U.S. Commercial Nuclear Power Plants

It has been over a decade since NUREG-1150 has been developed and a rebaselining of risk at this time is warranted. There have been several methodological advancements over the past ten years as well as an abundance of operating data. Further there have been changes in regulatory oversight, plant modifications, and changes in in-plant practices in maintenance, and operation. The current effort on risk-informing the regulatory process and the development of standards for PRAs, further underscore the need for this rebaselining. Methodological advances include faster running computer logic models, better modeling of common cause failures, new HRA models, improved Level 2 PRA models based on better understanding of accident progression phenomena, more consistent modeling on external events especially fire and flood, improved understanding on systems interaction due to instrumentation and control (I&C) systems, and improved off-site consequence models. In addition the effect of new maintenance and inspection techniques on plant operation should be reflected in the data used in PRAs. The effect of aging can now be better accounted for in PRA models, data, and methods. The body of knowledge on low power and shutdown risk should also be incorporated in the update of total plant risk. Some newer research challenges include reliability of digital I&C and distributed software control systems and reliability of passive components. These would have a bearing on PRAs for future reactors as well. In addition to updating the risk profile, a rebaselining study would also improve decision making, provide new focus for reactor research, and provide a benchmark for comparison with industry efforts, and enhance communication with the public.

8. Monitoring of Waste Storage and Disposal

The long term storage and disposal of radioactive waste, particularly spent fuel, is still problematic in the U.S. This is a public perception and political issue but there are still technical areas needing further research, such as development of appropriate monitoring techniques for various steps in the process to help ensure no adverse reactions or releases.

BNL has developed a neutron camera which has been used to monitor nuclear weapons in the former Soviet Union and which could be adapted for use in monitoring spent fuel stacks and waste container arrangements. Its use would help to avoid any potential criticality concerns should later degradation in the fuel and/or waste container barriers occur during transportation, processing, and long term storage.

II. ISSUES RELATED TO THE PBMR

1. Fuel and Fuel Coating Performance

The central issue to the passive safety of the PBMR is the ability of the coated fuel particles to retain fission product radionuclides. This, of course, includes both normal and off-normal operation. The assertion that adequate fuel performance can preclude catastrophic events will drive the reduction or omission of mitigation safety features, e.g., a robust containment, and the possible reduction of the exclusion and emergency planning zones. Hence there is a need to develop an independent capability to confirm fuel performance and fuel coating performance. This includes methods for beginning of life (quality assurance), as well as the means to project fuel performance to the end of life.

2. Non-Uniformities In Particle Bed Porosity

The flow through a porous media is a strong function of the porosity. Therefore, any non-uniformity in the porosity distribution in a particle bed subject to a pressure gradient will result in non-uniformities in the local fluid velocities. For a heat generating particle bed typical of the PBMR, this has many consequences. First, the porosity changes will directly result in changes (from the presumed average porosity) in the local power level due to changes in neutron moderation. Secondly, the consequent non-uniform flow distribution will result in changes in the local particle bed temperatures. This change in the local fuel temperature would then result in changes in the local power level via the negative feedback to reactivity. Thus, it is clear that a realistic porosity distribution (or several plausible distributions) should be considered in the thermal-hydraulic analysis of particle bed reactors.

Measurements of porosity distribution in a pebble bed of spherical particles in a hopper subject to vibrations have not only showed large variations in local porosity distribution after initial loading of particles, but also showed large changes in bed porosity distribution after the bed was subjected to vibrations. Clearly, detailed 3-dimensional flow prediction through a PBMR bed having inhomogeneous porosity distribution must be made if one is to predict a realistic peak fuel temperature during normal operation and various accident scenarios. It is therefore proposed that detailed (3-dimensional) thermal hydraulic analysis of flow through a typical PBMR be performed by using a modern computational fluid dynamics (CFD) code like FLUENT. This will allow us to simulate and study the effects of expected non-uniformities in bed porosity and local power production rate, and as well as the internal vessel structures within the particle bed.

3. Fuel Particle Inhomogeneities Within Spherical Pebbles Of PBMR

The difficulties in the manufacture of coated fuel particles, such as those proposed for the PBMR, are well known. Although estimates of failure rates, or the variation in the quality of manufactured particles, and hence also in the manufactured spherical fuel pellet, are difficult to estimate at present, it seems prudent to assume for now that one should also expect a certain inhomogeneity of fuel distribution within the spherical fuel pellet. This will result in an asymmetrical heat production within the sphere. Consequently, it becomes necessary to use a detailed 3-dimensional analysis to predict the temperature distribution within such a spherical particle subject to external flow. It is proposed that a modern computational tool like FLUENT be used for this analysis. Such an analysis, subject to parametric variations in fuel density distribution within the pellet, can bound the expected peak fuel temperature.

4. Development of General Design Criteria and Design Certification

The next generation of nuclear reactors, including the PBMR, can be radical departures from current and advance light water reactor designs. The current regulatory framework, as well as previous efforts for advanced reactor designs of the past, is inadequate to support the development of General Design Criteria and support a design certification. This can only be corrected through a complete revision of the relevant requirements.

5. Severe Accident Analysis for the PBMR

Probably the most severe accident that could be envisioned for the PBMR is the case of a leak in the system boundary that is subsequently followed by the ingress of air after depressurization. The flow field near the break would be very complex since the surrounding air would be entering the PBMR volume at the same time as the hot gases from the PBMR would be leaving the system. Clearly, a 3-D CFD code is needed to make realistic predictions of air infiltration into the PBMR under this accident scenario, and for evaluation of the resulting peak pebble temperature in the reactor. Furthermore, the CFD model must include tracking of more than one gas component (helium, oxygen, carbon dioxide, and nitrogen) and the graphite-air reaction as a function of graphite temperature and air concentration in the air-helium-carbon dioxide mixture. (Note that heat and carbon dioxide would be produced as a result of graphite-air reaction.) The FLUENT model developed under item (2) above would be used as a starting point for the development of this severe accident model.

6. Development of a combined thermal-hydraulics and neutronic code for the PBMR.

In order to study the response of the PBMR to various accident scenarios it is necessary to develop a code that is analogous to the RELAP5 code, which is used for conventional LWRs. The code should include all the elements (reactor vessel, turbine-compressors, turbine-generator, intercooler and precooler, etc.) of the PBMR power generating system. This code/model, together with models and analyses performed under items (2), (3), and (5) above would form an integrated set of tools for reviewing the PBMR design.

Technical Areas for Future NRC Research Idaho National Engineering and Environmental Laboratory

February 21, 2001

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At the request of Dr. Ashok Thadani, the Idaho National Engineering and Environmental Laboratory (INEEL) has identified the areas discussed below as ones that should be included in the NRC's future research program. The research topics have been placed in one of the following categories:

1. Pressing research issues for current nuclear facilities,
2. Improving regulations for current nuclear facilities, and
3. Regulatory framework for next-generation nuclear facilities.

While some of the issues identified below span more than one category, they are discussed under only one heading.

1. Pressing Research Issues for Current Nuclear Facilities

A. Organizational and Work Practice Influences on Risk and Reliability

Recent work in examining a sample of significant operating events in US commercial nuclear power plants has affirmed that the majority of these events are caused by latent failures. These failures are primarily rooted in maintenance and surveillance activities and to a lesser extent in technical knowledge and good work practices. The current generation of human reliability techniques has focused on active and not latent human errors. There is a need for a mature methodology to identify and address latent failures before they create conditions that can lead to an accident. Such a tool must be able to examine current as well as desired practices. It must also be able to assess the safety culture and identify deficiencies and stressors before they lead to failure events.

Cyber Vulnerability

With the advent of more sophisticated digital instrumentation and control systems and computer technology, as well as a more computer literate society, there is the potential that the US commercial nuclear industry is facing an ever-increasing vulnerability to cyber attack/intrusion. There is a need to identify and assess the vulnerabilities of computer-based systems (including interfaces and embedded firmware) in nuclear power plants whose reaction, failure, or malfunction due to insider or outsider cyber attacks would: (1) prevent the performance of the safety function of a nuclear power plant structure, system, component or equipment; (2) adversely affect the plant's physical security measures; or (3) degrade, impair, or prevent operation and continued electric power generation by the nuclear power plant.

Having identified the potential vulnerabilities, the postulated threats, and the critical assets associated with the information systems, an evaluation of the potential risk can be performed. Activities within this research program could include:

- identification of assets that are potentially vulnerable to cyber threats,
- network penetration testing using active scanning and other penetration tools to identify network, plant safety, security and operational vulnerabilities that might be easily exploited by a determined adversary,
- analysis to determine the influence/consequence of unauthorized access to critical facilities or information systems might have on system operations,
- identify and prioritize mitigation measures.

B. Motor-Operated and Air-Operated Valve Research

There is a number of unanswered research issues associated with the operability and aging characteristics of valves.

- Monitor Results of Ongoing In-plant Research and Implications to Valve Operability

  Review industry test data to better understand the effect preconditioning has on establishing and maintaining the operability of MOVs and its implication on operability assessment techniques. Understanding the effect of preconditioning is important when extrapolating the results of in situ testing to design-basis conditions to ensure the operability of an MOV.

- Monitor and Trend MOV Age-Related Degradations Using MCC Diagnostics

  Research is needed to determine the ability of motor control center (MCC) diagnostic tools to adequately monitor and trend age-related degradations of MOVs. MCC diagnostic systems are an emerging tool that is rapidly being embraced by the industry as part of a mix of test methods used for risk-informed IST. The adequacy and limitations of this technique to monitor age-related degradations have not been independently assessed.

- Limitorque HBC Gearbox Efficiency for AC- and DC-Powered MOVs

  Previous research has shown that the gearbox efficiency of a motor actuator can vary from published information as the load on an actuator increases and subsequently slows down. This variation typically reduces the output capability of an actuator and can impact the operability of an MOV if not given proper consideration. HBC actuators have not been evaluated, but efficiency concerns need to be understood to ensure that safety-related quarter-turn MOVs will be able to perform their design basis function.

- Accuracy of Industry AOV Diagnostic Equipment

  Research is needed to support industry efforts to determine the accuracy and adequacy of air-operated valve (AOV) diagnostic tools. The AOV Users Group is concerned about the accuracy of existing AOV diagnostic tools and they are reviewing means to independently validating the vendors accuracy claims. This effort is similar to the MOV diagnostic validation effort performed by MOV Users Group. At that time, industry and diagnostic representatives used the INEEL equipment operability laboratory and the results of the highly instrumented MOVLS to investigate the accuracy and adequacy of existing diagnostic test equipment.

- In Situ Monitoring of Stellite 6 Aging

  Research is needed to confirm in-plant behavior against existing laboratory testing on Stellite 6 MOV components with respect to corrosion and friction. This can be accomplished by instrumenting a valve in a typical plant environment using high precision diagnostics. Monitor the age-related changes in disc friction and correlate the results of corrosion and friction effects observed in the Stellite 6 testing to the response of in-plant equipment. This effort will verify that the separate effects testing provides bounding results and also ensures the continued operability of MOVs as they age.

- Operability Concerns with Actuators other than Limitorque

  Research is needed to determine whether other motor-operated valve actuators (such as Rotork) are adequately accounting for performance degradations that have been found to affect Limitorque actuators; for instance ac/dc operation at reduced voltage, actuator efficiency issues, elevated temperature operation.

- Stem Lubrication Capability as the Lubricant Ages

  Research is needed to determine whether the capabilities of typical stem lubricants degrade over time and result in an increase in the stem to stem-nut frictions. Such friction increases will decrease the operating margin of an MOV and could result in the valve not operating.

2. Improving Regulation for Current Nuclear Facilities

Best Estimate Plus Uncertainty Thermal-Hydraulic Code Development

The current state-of-the-art in thermal-hydraulic (T-H) analysis using system safety codes such as RELAP and TRAC provides best estimate calculations with no determination of any calculational uncertainty. Typically, regulatory T-H analyses and licensing codes are based on very conservative assumptions and inputs as defined in Appendix K. With such conservative analyses, uncertainty was not deemed important, the assumption being that the conservatism built into the analyses more than compensated for any uncertainties in the non-conservative direction. The use of best-estimate T-H analyses allows safety margins to be reduced, but also generates concerns about the uncertainty in the calculations.

Adding the capability to perform a complete uncertainty to current best estimate T-H codes would add a tremendous amount of calculational burden to an already computer-intense process. There is a research need to determine a workable solution to the problem of propagating uncertainty through the T-H calculations.

3. Regulatory Framework for Next-Generation Nuclear Facilities

A. Licensing Issues for Generation IV and Small Modular Reactors

Recent improvements in nuclear reactor designs have led to the development of numerous proposals for small modular nuclear reactors with a wide variety of engineering and safety features, including proliferation resistance. Small modular reactors (10-150 MW-electrical) have attractive characteristics for remote communities that otherwise must rely on shipments of relatively expensive and sometimes environmentally undesirable fuels for their electric power.

With the support of the US Department of Energy through the Generation IV Initiative and international interest in such designs, the NRC could face license requests for next generation, small modular reactor designs in the foreseeable future. An approach for addressing the regulatory issues associated with deployment of small modular reactors by the US commercial nuclear industry will be needed.

To monitor commercial nuclear plant performance, the NRC currently focuses on seven specific "cornerstones" that support the safety of plant operations in three broad strategic areas (reactor safety, radiation safety, and safeguards):

initiating events
mitigating systems
barrier integrity
emergency preparedness
public radiation safety
occupational radiation safety, and
physical protection.

A regulatory approach for small modular reactor designs should follow the current NRC approach, using adapted risk-based techniques and safety cornerstones to fit the concept of the small modular reactor. Ongoing efforts to risk-inform (10 CFR Part 50) should be incorporated as appropriate. Each NRC safety cornerstone objective should be addressed by the small modular reactor design. However, the approach for satisfying the cornerstone objectives may differ significantly from current light water reactor practices.

Any reactor design that differs from current light water reactor designs will require the development of a new set of licensing bases and criteria. Risk analysis methods should be used to develop a logical path for proceeding from the adapted cornerstones to specific design criteria and bases that can be used to design safety-related systems.

B. Thermal-Hydraulic Code for Non-PWRs/BWRS

The entire spectrum of current generation thermal-hydraulic (T-H) analysis codes was developed to specifically address pressurized and boiling light water reactor technologies. The Advanced Light Water Reactor designs produced and evaluated in the recent past required T-H code modifications to account for design differences such as passive safety systems and differing operating conditions. But for the most part, the principles, models, and physics of light water reactor technology still forms the basis for best estimate codes that are currently in use today. The next generation of reactors, such as the Gen IV designs that may come before the NRC for licensing will most likely not be light water reactors. Potential designs range from gas-cooled reactors to those that use lead-bismuth as a coolant. This will require a significant change in the fundamental principles, processes, and physics modeled by the codes. The calculational capability of the current water based PWR/BWR best estimate codes will not be adequate to perform safety analyses for these new designs. A new T-H system safety code for confirmatory and anticipatory analyses will be needed. Development of such a tool now will prevent delays in the licensing process for the first commercial next-generation reactor.

Contribution to NRC/RES White Paper on "Future NRC Directions" Los Alamos National Laboratory

March 7, 2001

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Licensing Reviews for New Reactor Concepts

The need to perform safety and licensing reviews of new and unique nuclear facilities in the coming decade will challenge both the intellectual and staff resources of the U. S. Nuclear Regulatory Commission (NRC). Two examples of potential review activities are the Pebble Bed Modular Reactor (PBMR) and the Accelerator Transmutation of Waste (ATW) facility.

The Exelon Corporation has recently met with the NRC with the objective of establishing a High Temperature Gas Reactor (HTGR) regulatory framework and associated policies by mid-2002. The effort to prepare the regulatory framework will include appropriate review and preparation of General Safety Requirements, a Format and Content Guide for HTGRs, and related detailed review plans. In addition, the NRC will need to identify the necessary tools and resources to support the regulatory review. Although the NRC licensed the Fort St. Vrain HTGR and was engaged in the review of large HTGRs several decades ago, a significant effort will be required to update the regulatory framework and acquire and apply the computational tools required to understand the complex physical behaviors related to safety. One such effort will be to incorporate risk-informed concepts into the licensing process.

Although there are some historical precedents and experience for the PBMR, only limited precedents exist for the NRC involvement in safety and licensability reviews for facilities used in the accelerator transmutation of waste. The ATW facility utilizes an accelerator to produce high-energy protons that are focused upon a spallation target. Neutrons are generated in the target and pass into the target to transmute long-lived radioactive materials. Various coolants for the blanket, e.g., lead-bismuth eutectic, sodium, and helium are under consideration. Each of these coolants offers unique challenges as will the need to understand the response of the ATW to various postulated accidents.

Phenomena Identification and Ranking

The licensing reviews for new reactor facilities described in the previous section will extend the staff and financial resources of the NRC. It will be necessary to ensure that the available resources are focused on those elements of each nuclear facility having the greatest potential to impact safety. A proven tool for supporting the logical and appropriate allocation of resources is the Phenomena Identification and Ranking Tables (PIRT) process. This process was first formalized as part of the Code Scaling, Applicability, and Uncertainty (CSAU) methodologies in the late 1980's as applied to the large-break loss-of-coolant accident in a Westinghouse four-loop pressurized water reactor (PWR). The PIRT process has since been applied to identifying and ranking phenomena and processes in applications as diverse as debris transport in PWR and boiling water reactor (BWR) containments, containment coatings failures, the response of high burnup fuel to various PWR and BWR accident scenarios, and burnup credit. The Advisory Committee on Reactor Safeguards has reviewed several PIRTs and both the process and results have been favorably received. Each PIRT that is developed can subsequently be used by the NRC as a tool to assist in the assessment of the adequacy of its computational tools and to ensure that testing programs, either planned or previously completed, provide the data needed to support licensability reviews.

Advanced Accelerator Applications (AAA) Requirements

High power, proton accelerator-driven spallation targets for nuclear applications are now at the forefront of advanced reactor research in the US, Europe, and Japan. Major projects in the US include the DOE Accelerator Production of Tritium (APT) program, and the Spallation Neutron Source at ORNL. A new initiative, Advanced Accelerator Applications supported through the DOE DP and NE offices is investigating the role of particle accelerators in waste transmutation. Planning in AAA is already underway for the construction of a high power demonstration accelerator/target facility.

In a letter dated August 2, 2000 to ANS President James A. Lake, the director of the Office of Nuclear Regulatory Research suggested several areas of potential interest for the NRC in this field, including:

- Development of a coordinated regulatory structure with the participation of agencies in addition to the NRC, although specific jurisdiction may need legislative clarification.
- Early engagement of NRC staff to understand the integrated system of high power accelerators, subcritical reactors, isotope separation and fuel fabrication facilities.
- Monitoring the research and development programs in order to ensure regulatory issues are identified and resolved early in the process. The NRC can help in reducing cost and time delays by bringing previous experience to the community, particularly in identifying needed confirmatory research. Of particular concern are novel spallation target concepts now under consideration, such as Lead-Bismuth-Eutectic technology.
- Modifications to nuclear and thermal-hydraulic codes and data (materials properties, neutron cross sections, decay heat tables, etc.), needed to analyze accident situations.
- Adaptation of safety analysis methodology now in use for commercial power plants to the subcritical reactor/spallation target envisioned by AAA.

The extensive research program carried out by APT made advances in several of the technical areas, however the specific focus of that program did not encompass subcritical reactor technology. As an example, the high power proton materials irradiations carried out at the LANSCE accelerator indicated that the onset of transition and failure properties are closely related to the buildup of hydrogen and helium from the spallation process, which occurs much faster than in conventional reactor experience. Maximum yield in proton irradiated materials can occur at a few dpa rather than the 10's of dpa now seen in existing reactors. This has necessitated the formulation of structural design criteria and stress allowables specific to this environment. In transmutation activities involving a spallation target surrounded by a subcritical reactor, the proton effects will not be confined solely to the target, but also extend into nearby fuel elements. NRC guidance to the AAA community is needed to ensure that current methodologies for materials research meets appropriate regulatory standards, and if new requirements must be met, they need to be formulated in time for the design of the demonstration facility.

Modifications to the TRAC code have already been made at Los Alamos to incorporate lead-bismuth and gas coolants, and further changes can be made depending on the final target choice. In neutronics, lack of high-energy multigroup data libraries prohibits the use of some standard reactor analysis tools. The lack of data is not the only problem; difficulties in handling the very forward peaked distributions resulting from the spallation process have never been adequately solved in deterministic methodology, and incorporating charged particle scattering is an unsolved problem. As a result, Monte Carlo techniques are in exclusive use for spallation target design and analysis, and above existing data libraries energy limits, on-line physics modules calculate interaction probabilities directly. This methodology, long in use in the high-energy physics community, needs regulatory review and approval. Benchmarking activities carried on in the APT program have made inroads into validation of codes for neutronics performance, energy deposition and decay heat, however they have been carried out largely without regulatory supervision. Review and advice at this point is needed.

Within the US, unique facilities, e.g., LANSCE, exist for continuing materials and benchmarking research. A Materials Handbook and structural design criteria specifically addressing the accelerator community have been widely distributed. Originally based on APT work, these should be extended to transmutation issues. Work extending the MCNP code to all particles and all energies (MCNPX) and extending existing data libraries to 150 MeV has also been carried out under the APT program. Due to the nature of the APT project, actinide issues in materials and the codes could not be addressed. Adaptations to the TRAC code have been made, and other safety analysis tools have been used, although some extrapolations have been made since available data is inadequate. For example, the existing base of biological dose equivalent factors is inadequate to cover the spallation product range. Safety-specific methodologies must also be adapted to the new environment, particularly for the SIMMER-III code. Support from NRC Research programs is needed to fully adapt the analysis tools and to validate existing and proposed benchmark and materials research for the new applications.

NRC/RES White Paper
Lawrence Livermore National Laboratory

February 27, 2001

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There are substantial nuclear opportunities and challenges on the horizon, and that horizon is not that far into the future. In the cases of a spent fuel repository and pebble bed gas reactors, it is upon us. Will we be ready to support the licensing and safety oversight of these emerging entities, or will voids in technical understanding yield an uncertain climate for decision making?

Current Nuclear Systems

The existing nuclear fleet is quite mature. Many substantial safety issues have been addressed, albeit in a very conservative way. This conservatism is understandable in the absence of perfect knowledge. However, as time has progressed, our understanding of systems and material properties has increased, and the need for such conservatism is reduced. In some cases (e.g., snubbers) safety was degraded through an incomplete appreciation of a systems approach to safety assessment. The willingness to reduce this conservatism must be rooted in a risk-informed approach, and the technical basis for this approach must have firm research underpinnings. The requirement for containment spray in the AP600 indicates we are not there yet. In addition to research needed to risk-inform regulations, the need to understand aging impacts on systems, structures, and components is necessary.

Future Nuclear Systems

The bulk of NRC/RES must be focused on emerging nuclear systems, especially since these systems represent a departure from what NRC has devoted substantial research to (i.e., light water reactors). These systems included the proposed nuclear waste repository (in particular, the engineered barrier system), pebble bed gas reactor, and potentially liquid metal systems. In addition to these civilian systems, the disposition of weapons material, especially plutonium, will have some nuclear facilities under NRC purview involving materials and systems that NRC has had little historical involvement. In my view, the following areas are especially important for anticipatory research:

Materials

The understanding of new (e.g., metal alloys) and novel (composites, nanolayers & multilayers) materials will require fundamental research. The long-term performance of nuclear waste containers will be an issue during repository licensing; will we have the underlying material research basis? The traditional "defense-in-depth" approach to fission product release to the environment (fuel cladding > RPV > containment > environment) will be challenged by the new fuel forms being envisioned. In the case of triso-coated fuel particles, will we be able to internalize a similar "defense-in-depth" approach when all the layers are associated with the fuel form (multilayer #1 > multilayer #2 > multilayer #3 > coolant > environment)? Although we have experience with sodium liquid metal systems, the emergence of systems focused on lead or lead-bismuth will require understanding of material performance and corrosion with these coolants. In some system proposals, life-of-core (15 years) is being advocated for economic and non-proliferation objectives. Will materials and systems be up to the task for these noteworthy objectives?

Instrumentation & Control

The computer and Internet revolution has largely bypassed nuclear power plants. Multi-megawatt natural gas plants are routinely managed by a small handful of people using computer-based controls. The ability (or current inability) to upgrade controls and instrumentation to contemporary standards hinges greatly on the regulatory basis. Modern I&C systems enhance reliability, lower cost, and (potentially) enhance safety through predictive maintenance and automated response to safety challenges. Clearly the aircraft and automobile industries are placing computers in safety-critical roles. Research is needed to address any nuclear-specific issues that may relate to computer-based control over plant operations.

Systems-Level Modeling

The advances in computing power over the last 25 years have been phenomenal. The ability to model complex, multi-scale systems with high fidelity is within our grasp. Ideally, one could model atomistic material behavior in a radiation environment through bulk material performance over 50 years. These various material models could, in turn, be integrated into a complete model of systems, structures, and components. The addition of coolant behavior, thermal and neutronic performance is required. Lastly, the instrumentation and control system would be modeled and integrated, yielding a "virtual reactor" that could be constructed, operated, maintained and stressed as fast as supercomputers could run. The ability to perform parameter studies would be invaluable. Of course, there needs to be a balance between computer models and underlying experimental basis, but this area offers tremendous research opportunity and value to the nuclear community.

Some Technical Areas Important to the U.S. Nuclear Regulatory Commission's (NRC's) Research Program

Oak Ridge National Laboratory

April 10, 2001

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The areas shown below have been identified by experienced staff members at the Oak Ridge National Laboratory (ORNL) as ones that should be included in the NRC's forward research program. This list is not meant to be complete, rather it highlights topics felt to be of sufficient importance for inclusion by NRC. Most of the attention here is on current operating power reactors. The results would contribute directly to NRC's goals of providing enhanced safety, reduction of unnecessary conservatism, and risk-informing specific rules. While additional attention is required to comprehensively identify the research needed to support certification or licensing of advanced reactors, a few thoughts are included in Section III below.

The guidance for formatting this list asked that issues requiring research be placed under the following three categories:

1. Pressing research issues for current nuclear facilities (confirmatory research),
2. Improving regulations for current nuclear facilities, and
3. Regulatory framework for next-generation nuclear facilities (anticipatory research)

Even though some research areas identified below are of a continuing nature, they have been placed under one of these three headings.

I. RESEARCH CATEGORY 1: PRESSING RESEARCH FOR CURRENT NUCLEAR FACILITIES

A. Assess Impact of Updated Technologies on Regulations for Primary System Components under Normal and Off-Normal Conditions

Structural Integrity of RPVs and Other Primary System Components

The most significant assessment item in the near term is completion of the thorough reevaluation of current rules for ensuring integrity of reactor pressure vessels (RPVs) under Pressurized Thermal Shock (PTS) scenarios. Updated analysis models, computer codes, material data, statistical treatment of uncertainties, and system characterization contribute to improved understanding of margins and uncertainties in current rules. For example, potential reductions of the margins and uncertainties associated with flaw density and distribution, fracture toughness and radiation-induced embrittlement of the RPV materials, can have significant effects on operation of the plant, especially in the case of license extension. This assessment process can also evaluate alternative approaches that may provide relief where unnecessary regulatory burden (conservatism) is identified. The experience gained from the PTS reevaluation should also be applied to other areas where technical advances have occurred since the current regulatory practices were established. This includes examining other transient situations, such as those associated with start-up/shut-down cycles, and components exposed to conditions that could lead to fatigue and/or cyclic crack-growth damage (e.g., vessel internals, outlet piping components, and steam generator components).

B. Safety Evaluations for Digital Instrumentation and Control Systems

Safety-Related Digital Instrumentation and Control Systems

The operation of current nuclear power plants through license renewal period, coupled with obsolescence of existing analog components, increases the likelihood of the use of digital technology in safety-related instrumentation and control (I&C) applications. In addition, Generation IV reactor systems will predominantly employ digital technology for I&C applications. While digital technology can provide increased functionality, self-checking capabilities, and high integrity components, its performance and reliability characteristics are not understood as well as they are for their analog predecessors. As a result, continued research is needed to enhance the technical basis for evaluating safety-related I&C upgrades using digital technology (including commercial-off-the-shelf equipment) or next generation digital systems. Such research supports the goals of enhanced safety and reduction of unnecessary regulatory burden. The former goal is addressed through further reduction of the possibility for failure of function or unintended function. The latter goal is addressed through better understanding of the safety characteristics of digital technology and the elimination of uncertainties leading to unnecessary conservatism.

Potential areas for research into safety-related applications of digital I&C include:

- reliability estimation for hardware/software systems,
- software qualification (e.g., functional certification),
- benefits/consequences of software diagnostic approaches,
- reliability and robustness characteristics for communications architectures (e.g., wireless networks),
- in-service monitoring and prognostics for digital systems, and
- environmental effects (e.g., power quality, electrostatic discharge).

Surveillance and Diagnostics for License Renewal Periods

Nuclear plant operation during license renewal periods calls for the use of methods to detect age-related degradation of key reactor and nuclear steam supply system components (e.g., core internals, pumps, steam generators). In the 1980s, NRC sponsored confirmatory research to evaluate on-line surveillance and diagnostic methods for detection and diagnosis of plant system degradation. The results of this research provided methods to address several generic problems. Since then, utilities in other countries have made extensive use of on-line monitoring and diagnostic methods, and significant progress has been made in the use of new automated surveillance methods, including such technologies as neural networks and wavelet signal processing. There is now an extensive array of commercially available surveillance technologies, and new methods are continuing to be developed and employed in foreign reactors. Research into this technical area would supplement the technical basis for evaluating license extension issues by assessing foreign experience with monitoring and diagnostic technologies and identifying emerging methods that may be employed in the U.S.

C. MOX Fuel PIE Testing

In regards to research needs concerning the use of high-burnup and advanced fuels in current reactors, it is important to note that the DOE-sponsored Fissile Materials Disposition Program is currently conducting at ORNL the post-irradiation examination (PIE) series for the weapons-derived mixed-oxide (MOX) fuel irradiated at INEEL. While the fuel pins examined in the present studies are from short rods, full-length fuel rods irradiated in the Lead Test Assemblies (LTAs) are scheduled for future PIE at ORNL. Plans for upgrade of the ORNL hot cells to handle full-length fuel rods are being implemented. To minimize costs, NRC might consider collaboration with the DOE program via existing MOUs.

II. RESEARCH CATEGORY 2: IMPROVING REGULATIONS FOR CURRENT NUCLEAR FACILITIES

A. In-Reactor Technology

Enhanced Technologies for Reactor Pressure Vessel Integrity Evaluations

As nuclear plants operate deeper into their initial 40-year license period and potentially beyond, the need grows to verify that RPV data and regulatory rules are applicable to this time regime and the cumulative environmental exposure. Accordingly, regulatory instruments and analysis tools need to be regularly evaluated against advancements made in thermal hydraulics, fracture mechanics, PRA, material behavior modeling, etc. With respect to fracture mechanics, advancements currently under development are introducing nonlinear material response models applicable to constraint and size variations. These advancements will lead to more capable approaches for probabilistic RPV integrity assessments. Innovative representations of fracture properties of steels, such as the Master Curve approach, hold potential for refining margin definition. Knowledge is increasing on how material properties degrade with time, temperature, radiation exposure, and operating atmosphere. Thus, a strong research endeavor needs to continue as plants age to gain applicable data for characterizing known factors and for detecting evolving factors that can influence integrity of components. The most critical area is radiation embrittlement of steel components, including RPVs. Research needs to continue on phenomenological and mechanistic scales to understand and predict embrittlement effects due to long-term neutron exposure. The most significant issues relative to impact on the current regulatory process are: (1) material variability and surrogate materials, (2) high fluence, long irradiation times, and flux effects, (3) master curve fracture toughness, (4) through vessel wall attenuation, (5) high-nickel welds, (6) modeling and microstructural analysis, and (7) post-annealing embrittlement.

Characterization of Fluence for Pressure Vessel and Reactor Internals

Computational methods and tools to characterize the neutron fluence in RPVs and reactor internals need to be re-assessed given the extended lifetime of PWRs and recent findings that indicate BWR shroud components and internals may be susceptible to material damage due to neutron fluence. The current methodologies for analyzing the fluence in PWR RPVs may not be adequate for the added geometrical complexity, distance from dosimeter locations, and varying conditions found in a BWR. In addition, use of mixed-oxide (MOX) fuel in PWRs may affect the spectrum such that RPV damage correlations developed for UO_2 fuel may not be adequate for cores loaded with a substantial amount of MOX fuel.

B. Studies Supporting Reactor Fuel and Out-of-Reactor Facilities

Nuclear Safety Computational Analyses Capabilities

Maintenance, enhancement, and application of nuclear safety software in the areas of criticality safety, radiation protection, and spent fuel characterization should be an important part of NRC's ongoing research program. Besides their role in investigating safety-related issues, research must increasingly rely on computational approaches to augment the definition, interpretation, and utilization of new and/or existing experiments and measurements. NRC research needs to continue to provide and employ cutting-edge computational analyses to ensure that the best possible information is obtained from existing experiments, and that new experiments are properly defined and justified. Experimental data has been and will continue to be an important component of research; however, the loss of experimental facilities and the rise in cost for obtaining experimental data are often impediments to resolving important technical issues. Computational approaches can now utilize advanced sensitivity and uncertainty methodologies to help demonstrate that existing experimental data are, or are not, satisfactory to address open issues. These approaches are now being used to help address the paucity of experimental data in areas such as burnup credit and source term uncertainties for high burnup fuel. Expanded use of the sensitivity/uncertainty approaches to other application areas should be considered.

Review and Assess Computational Methodologies for Advanced Fuel and Reactor Types

Work is needed to review and investigate the adequacy of computational methods and data available to assess the processing, fabrication, reactor utilization, transport, and storage of advanced fuel concepts such as weapons-grade MOX material and pebble-bed fuel spheres. The capabilities and limitations of existing software and data may not be adequate to properly handle the needs of the agency for confirmatory review. The research should recommend, enhance and/or modify codes to be used by NRC as appropriate.

Research for Risk-Informing Non-Reactor Facility Assessment

Research is needed to aid in the effective utilization of integrated safety analysis and human factors research in the safety assessment of out-of-reactor facilities. How can PRA be effectively used? Work in the computational sciences has also been done to provide models that enable assessment of human performance and use of procedures in safety environments. The goal of the research would be to provide risk-informed modeling capabilities that can support the assessment of facility safety.

C. Enhanced Technology for Safety Evaluations for Boiling Water Reactors

A number of specific areas of BWR technologies merit priority work to develop improvements. These include the following:

1. Assess the impact of higher decay heats associated with the plant power uprates to determine the effect on the timing and severity of severe accidents. Is there any significant effect on the progression of events? (For example, would containment failure be predicted after the power uprate but not before?)

2. Determine whether or not the industry MAAP code is capable of correctly calculating the behavior of partial-length fuel, now being introduced into BWRs. Does the use of such fuel affect the design criterion that core melting will not occur as long as the core remains flooded to 2/3 core height?

3. Access the consequences if a BWR core is uncovered, the control blades melt, and the standing fuel within the core is then recovered. (This is one possible branch of the Station Blackout event tree, with late recovery of electrical power.) This event has been considered before, but a reliable estimate of the power generation in the bare core has never been obtained.

4. Review the case for closure of the Mark I liner failure issue, given that the results of the Individual Plant Examinations (IPEs) indicate that BWR vessel failure at high pressure is NOT excluded. (NUREGs 5423 and 6025 indicate that liner failure is probable if vessel blowdown is at high pressure, even with the recommended fix of water overlying the floor.)

5. Previously, the NRC decided not to consider high-pressure melt ejection (HPME) events for BWRs because it was thought that such high-pressure blowdown would not occur. With the results from the IPEs (see previous Item) indicating otherwise, it seems reasonable that the research should be undertaken to consider HPME in BWR containments.

D. Safety Evaluations for Scenarios Involving Fuel Rods Exposed to Air Environment

Credible scenarios where fuel rods might be exposed to air environments need to be analyzed to ensure that both the scenario initiators and the phenomena associated with potential fission product release are well understood. These scenarios should include reactor operating events as well as any plausible spent-fuel storage situations. Computational modeling should be coupled with existing experimental data to further study current questions on fission product release and establish an understanding of the level of uncertainty associated with event probability and consequences. The NRC research program should contain sufficient flexibility to accommodate the appropriate type and number of experiments should they be deemed necessary.

E. Risk-Informing Digital I&C Regulation

Given the goal of risk-informed regulation, new tools are needed to assess I&C systems. PRA evaluations typically do not include I&C system details. An assessment tool should be developed using detailed I&C models to automate failure mode and effects analyses (FMEAs) and to perform relative risk assessments to identify safety-relevant research areas.

III. RESEARCH CATEGORY 3: REGULATORY FRAMEWORK FOR NEXT GENERATION NUCLEAR FACILITIES

A. Some General Comments Concerning Advanced Reactors and Advanced Nuclear Systems

For the proposed advanced nuclear power reactors and for the proposed non-traditional nuclear systems, neither of which have been certified nor previously licensed or in some cases even reviewed by the NRC, the applicability of prescribed regulatory rules/framework must be established. The research required to confirm the adequacy of the regulatory framework and the associated rules and implementing guidance depends on the reactor or the nuclear system concept.

This new framework must address items such as qualification of fuel, accident scenarios, load characterizations, materials selection, material databases, and analysis methods as well as the rules and criteria that set limits on performance factors that affect safety. Assessing the embrittlement characteristics of the proposed Generation IV structural materials is also critical. Some advanced reactor concepts involve features (e.g., fuels, coolants, and thermal environments) that are characteristically different from current water-cooled reactors. For example, some have operating temperatures that are sufficiently high to place portions of primary system components in the creep (time-dependent deformation) regime. Other advanced nuclear concepts have radically different operating regimes such as accelerator-driven sub-critical reactors that obviate the importance of inherent reactivity feedbacks. Other prominent areas of difference include fuel integrity, source term specification, and severe accident scenarios. Regulatory rules and associated computational tools must reflect the specific differences; however, much can be learned by identifying and addressing the fundamental safety functions that underlie the regulatory framework for licensing the current generation and advanced light water reactors. By focusing on the key safety functions as implemented in the General Design Criteria of light water reactors, analogies can be drawn between the current set of regulatory requirements and the appropriate set of regulatory requirements that can be applied consistently to the next generation of nuclear systems. Thus, both anticipatory and confirmatory research is needed to ensure the adequacy of potential approaches, models, and codes in light of some fundamental safety functions that must be accomplished or accommodated no matter how different or unique the nuclear application is from what is considered traditional..

In the remainder of this section, we will focus on the regulatory research needs of the following items:

- High-temperature Material Data and Structural Integrity Assessment Methods
- Pebble Bed Gas-Cooled Modular Reactor (PBMR)
- Accelerator-Driven Systems (ADS)

B. High-temperature Material Data and Structural Integrity Assessment Methods

Several advanced reactor types operate at temperatures sufficiently high to induce time-dependent creep strains and/or stress relaxation in primary system components. In the presence of simultaneous cyclic loadings, this time-dependent material response can interact synergistically with fatigue to produce creep-fatigue damage. Limiting this damage to acceptable levels for the assurance of structural integrity has become the primary challenge in the design of elevated-temperature components, both nuclear and non-nuclear, that undergo cyclic loading. Developers of Subsection NH, *Rules for Construction of Nuclear Power Plant Components*, of Section III of the ASME Boiler and Pressure Vessel Code have wrestled with this problem, with only partial success, for more than 30 years. Models for predicting the complex material response and design criteria for limiting the damage are still plagued with uncertainties, which are recognized worldwide. Data and models should be generated to allow NRC staff to independently evaluate and judge the conservatism built into the calculational methods and design criteria. The needed experiments consist of basic high-temperature materials tests, using relevant test histograms, and small, relatively simple, confirmatory structural tests, aimed at providing data for assessing the overall life-assurance process. Such test data, coupled with the knowledge of the degradation or change in materials due to irradiation and aging will be needed to support any license-application action for proposed high-temperature reactor systems such as the PBMR, the French-proposed gas-cooled fast reactor, the Russian-proposed heavy-liquid-metal-cooled reactors, and the molten-salt-cooled reactors.

C. Comments Concerning Regulatory Research for the Pebble Bed Gas-Cooled Modular Reactor (PBMR)

The PBMR is one advanced reactor concept that is currently receiving considerable attention in the trade press and by the utility industry. Previous NRC reviews of coated particle fueled reactors have been of the annular spine fuel elements (Peach Bottom Unit 1) and the extruded-compact element in prismatic graphite blocks (Fort Saint Vrain, the large HTGR, and the MHTGR). The PBMR uses coated particle fuel in graphite pebbles similar to those used in the AVR research reactor and the Thorium High-Temperature Reactor (THTR) in Germany. However, the PBMR will operate at the higher temperatures and gas pressures needed for a gas-turbine power conversion cycle and will use only low-enriched uranium oxide fuel particles as opposed to the high-enriched uranium-thorium oxide particles or the high-end low enriched uranium oxide fuel particles proposed for the MHTGR. The PBMR retains the high-temperature ceramic core and the TRISO-coated fuel particles characteristic of the MHTGR and relies upon the passive cool-down safety feature to preclude fuel temperatures from reaching the point of where the particle coatings of pyro-carbon and silicon carbide lose the capability to retain most fission products. The proponents of the PBMR propose relying upon the core's passive safety features to avoid requiring a containment. The principal technical issues of importance to safety remain the same as or similar to those for the MHTGR:

- The effectiveness of coated particle fuels to retain fission products during both normal operations and accident conditions (core heat up and water ingress although the latter is less likely except during shutdown as occurred at Fort. St. Vrain).
- The quality assurance and quality control of both fuel kernal fabrication and particle coating processes to minimize coated particle external contamination with fissionable material, the number of particles with failed coatings, and the number of particles with degraded coatings that could fail during heat up transients.
- The integrity of vessels for the reactor and the power conversion system and of the interconnecting vessels or piping. The MHTGR would have operated at lower temperatures and pressures and therefore was proposed to use carbon steel vessels based on LWR experience with a special ASME Code case for the higher temperatures that would be experienced during a core cool-down event. Due to higher operating pressures and temperatures, the PBMR vessel will likely use a ferritic stainless steel similar to 2-1/4Cr-1Mo, 9Cr-1Mo, or a foreign equivalent for which an acceptable set of Code cases must be developed and reviewed.
- The integrity of penetrations into the vessels and the significance of penetration failures both for the blow-down of circulating and plated-out fission product inventories as a potential source term and for air ingress events leading to possible conditions for graphite oxidation. The ORNL Graphite Reactor Severe Accident Code (GRSAC), which has been derived from the ORECA/MORECA codes used in previous NRC analyses of HTGR accident scenarios, incorporates graphite oxidation models that have been compared against the Windscale accident.
- The integrity of the core barrel, in-vessel thermal shields, metallic core support floor, and hot duct due to normal, cyclic, and transient thermal conditions and seismic events (assuming also that Alloy 800 or a foreign equivalent austenitic alloy is the primary material of choice as it was for the

MHTGR). The integrity of these components must be assured so that configurations assumed in design basis and beyond-design-basis accident analyses are maintained and that active cooling of the core is always an option for the operators in recovering from an accident or event.

- The integrity of in reactor ceramics (alumina or fused silica) used as insulation on the core support floor. The integrity of these components must be assured so that configurations assumed in design basis and beyond-design-basis accident analyses are maintained and that active cooling of the core is always an option for the operators in recovering from an accident or event.
- The integrity of reflector graphite, core support graphite, and other large graphite structural support components subject to combinations of loading, irradiation, thermal cycling, and possible oxidation environments if restart occurs following a post-shutdown water or chemical ingress. The integrity of these components must be assured so that configurations assumed in design basis and beyond-design-basis accident analyses are maintained and that active cooling of the core is always an option for the operators in recovering from an accident or event.
- The performance and integrity of fuel pebble and reflector pebble graphite that must survive irradiation, thermal gradients, and surface stress while allowing the multiple passage of pebbles through the core. The long-term stability of the graphite ball inner reflector must be assessed in particular. The review and evaluation of German data are needed first with independent assessment of the results of relevant tests and operating experience.
- The performance of carbon-carbon components, such as proposed for canning material for high-temperature control rods, and the potential degradation effects of radiation, undetected water or chemical ingress, or elevated boron oxide contaminants in the boron carbide.
- The effects of graphite grit or hydrogen (if titanium is used in the turbines) on the blade shedding potential of the compressor or power turbines and the impact of blade shedding events.
- Aging effects on the surface emissivity of components (reactor vessel, core barrel, and reactor cavity wall) important-to-safety in assuring effective heat transfer during passive core cool-down events.
- The criteria to be used to define quality, inspection and test, and technical specification requirements for important-to-safety equipment such as passive cool-down surfaces and piping systems and for the back-up primary coolant circulation and cooling systems and their associated power sources. Previous reviews of such systems for the MHTGR and the AP-600 may serve as a guide.
- Independent and confirmatory analyses or tests of passive cool-down performance by the fuel and by key structures of the reactor system. GRSAC needs to be modified to simulate PBMR accident scenarios.
- The assessment of the issues of no containment (confinement only) versus a vented-filtered containment versus a sealed containment.
- Potential dose rates to maintenance personnel due to fission product plate-out on primary coolant system components (especially radioactive silver on the turbine blades).
- The paucity of high-quality and relevant critical experiment data or core physics start-up test data for graphite-moderated systems. Both the effects of xenon-135 and the thermal fission resonance of plutonium-239 can cause the at-power power reactivity coefficient in low enriched uranium systems to be near zero or even slightly positive at normal operating conditions. However, in the event of a loss of cooling, the plutonium-240 thermal capture resonance that is effective at higher temperatures will slow the power rise, and the concurrent decay of iodine-135 into xenon-135 will bring the core sub-critical.

D. Comments Concerning Regulatory Research for Accelerator-Driven Systems (ADS)

The ADS concepts for source-driven sub-critical reactors have received much attention in Europe and Japan as well as the United States as a means to transmute nuclear wastes in lieu of using a critical reactor such as that proposed previously for the Integrated Fast Reactor (IFR). However, no serious or credible proponent exists for a commercial application. ADS do deserve some attention for three reasons:

1. Currently, the *Atomic Energy Act of 1954*, as amended, does not provide a legal basis for the NRC to regulate or license the production, use, and disposal of accelerator-produced radioactive materials (ARM) so that regulatory responsibility continues to devolve upon the states for this type of "orphan source." It is understood that the Commission is actively considering requests to seek Congressional authority to regulate ARM as well as naturally occurring radioactive material (NORM). It is unclear, however, as to how the requested authority would be phrased in the enabling legislation, and the selection of wording may have larger implications. Would it be NRC's intent to have to license all ADS under 10 CFR Part 50 whether or not such systems use actinide isotopes?
2. If NRC decides to regulate ADS that only include sub-critical reactors, how would the definition of "nuclear reactor" in 10 CFR 50.2 have to be changed? Would the definition have to address the issues of "safeguards" and "material accountability?" Would the definition have to address the minimum amount of irradiation needed to change fuel material classified as "source material" under 10 CFR part 40 to "special nuclear material" under 10 CFR Part 70? Would the definition have to include the amount of irradiation sufficient to cause irradiated material to be classified as "spent fuel?"
3. ADS lack "general design criteria." The Department of Energy has never developed these for its accelerators, and it is not clear that ADS can be declared to comply with General Design Criterion 11. How would NRC address a reactor that cannot survive an ATWS? What other design criteria would have to be modified (possibly to account for risk-informed regulation) so that Criterion 11 can be bypassed in a defensible manner in licensing ADS in the future?

Technical Areas Important to the U.S. Nuclear Regulatory Commission's (NRC's) Research Program

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February 22, 2001

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There has been a good deal of discussion related to the need for NRC performing anticipatory versus confirmatory research. The latter is viewed as serving the immediate needs of NRR and NMSS, whereas the former is forward looking and may not have immediately obvious applications. While it is clear that the NRC research program should support day to day licensing activities, it is also clear that, in the absence of forward looking activities, the agency could find itself without answers when new issues or unanticipated problems arise. For example, the development and application of Probabilistic Risk Assessment technology and the subsequent evolution of risk-informed and performance-based regulation would not have taken place in the absence of anticipatory research. Consideration of the safety issues associated with new reactor designs is an obvious example of needed anticipatory research, but one that is of little interest to the power reactors that are currently being operated.

Experienced staff members at Pacific Northwest National Laboratory (PNNL) have identified the research areas listed below as topics that should be included in the NRC's research program. This list is not meant to be complete, but it attempts to highlight topics felt to be of sufficient importance for inclusion by NRC. Conducting this research would contribute directly to NRC's goals of providing enhanced safety, reduction of unnecessary conservatism, and risk-informing specific rules.

Nuclear Materials-Related Research

<u>Void Swelling in PWR Austenitic Near-core Internals</u>

While no substantial swelling is anticipated in austenitic steels in BWRs because of lower exposures and temperatures, certain less well-cooled portions of the Westinghouse and possibly Combustion Engineering baffle-former designs (in re-entrant corners) will experience temperatures and neutron exposures during a 40 or 60 year lifetime that may produce more than 10% swelling. At ~10% swelling the tearing modulus of stainless steel goes to zero when physical insults occur at low temperatures following shutdown. This introduces a well-known type of severe embrittlement often seen in fast reactors. In fact, this degradation mechanism became the life-limiting criterion for fast reactor fuel pins. The implications of such previously unanticipated embrittlement for PWRs has not yet been explored.

Research is needed to focus on the detection and prediction of void swelling at much lower doses to properly respond to re-licensing applications. Non-destructive examination techniques will need to be established and verified to quantify the ability for detecting cavities in stainless steels and monitor their growth. Modeling of void swelling will require a detailed assessment of local radiation dose conditions (flux and thermal-to-fast neutron ratio), sensitivity of the bulk alloy composition to swelling and the kinetics of void growth. It is important to note that swelling in the PWR baffle plates may cause high local stresses and promote damage within the core baffle-former structure. For example, high stresses produced on the baffle-former bolts can lead to stress corrosion cracking as has been seen in several European PWRs. Overall, building expertise in void swelling may be an essential requirement for the NRC to address re-licensing issues and in the assessment of advanced reactor designs such as gas-cooled fast reactors. This issue is an international concern.

Environment-induced Cracking in Nuclear Components
Corrosion and stress corrosion cracking have consistently limited light-water reactor performance for more than 30 years. Unexpected component failures, such as the recent stress-corrosion cracks identified at V.C. Summers, continue to have significant safety implications. Environment-induced cracking continues to be observed in core regions (shroud and top-guide structures in boiling-water reactors, and baffle bolts in pressurized-water reactors) and in many critical out-of-core regions of the plant (steam generator tubing, primary water piping systems, nozzles and valves). All of these examples result from an iron- or nickel-base stainless alloy exposed to high-temperature water. Although current understanding of degradation phenomenology has improved, the mechanistic causes for crack initiation and advance have not been identified in most cases. This limits the ability to predict environmental cracking with confidence and makes it nearly impossible to accurately assess the effectiveness of material changes. Mechanistic understanding should be established to address the implications of individual component failures, materials repair or replacement, aging and general life extension issues, and potential new reactor design issues. Recent experimental advances have revolutionized the ability to assess stress corrosion cracking. For example, nanoscale characterization of the local structure and chemistry at crack tips has identified specific corrosion reactions driving crack growth. Such techniques make it possible to determine the mechanisms controlling degradation and can form the basis for material and component assessment in reactor environments. As noted for void swelling, environmental-induced cracking is also an international concern with a potential for important international collaborations.

NDE-Related Research

<u>Non- Destructive Evaluation of Nuclear Components</u>
Given the bath tub curve of failure, as nuclear power plants enter the later stages of their licensing periods there may be more components that fail in these plants. This may be reflected in the recent problems at PWRs such as V.C. Summers (inconel nozzle to piping weld), Oconee (inconel CRD penetration), and Indian Point (steam generator tubing). A number of operating plants are requesting license extensions for an additional 20 years of operation, which will lead to additional aging and degradation that must be managed. There is also pressure on utilities to reduce operating costs and increase their efficiency. A keystone to the defense-in-depth of nuclear power plant operation is the nondestructive examination (NDE) conducted as part of the inservice inspection (ISI) program to reliably detect and accurately size degradation before it leads to leaks or challenges the structural integrity of components. The aforementioned problems at PWRs resulted from not reliably detecting degradation and, when it was detected, the inability to accurately characterize it.

Extensive work by RES has resulted in significant improvements to ISI methods through risk informed ISI programs (focus NDE on those nuclear components that contribute the greatest risk) and ASME Boiler and Pressure Vessel Code Section XI Appendix VIII (performance demonstration of NDE personnel, equipment and procedures). Even with these changes, there are substantial improvements still needed.

There are classes of NDE inspection problems that require additional research. RES has taken the position of reviewing the research the industry performs, but in some cases the industry is not planning on conducting the needed research. In fact, the industry is working within the ASME Code to implement changes eliminating some inspections because of inspection reliability difficulties.

One class of difficulties that is being addressed in this manner is the inspection of coarse grained materials, which includes centrifugally cast stainless steel, static cast stainless steel, dissimilar metal welds and single sided austenitic steel. These are all very challenging inspections, and solid research programs will be required to identify solutions. In some cases the industry is taking the approach of Appendix VIII performance demonstrations, which are being conducted on a best effort basis for current technology. In other cases, the industry is not performing any research, and is not supporting development of any performance demonstration requirements such as those in Supplement 9 of Appendix VIII. There is a need for RES to be more proactive in conducting limited research that indicates to the industry that solutions to the inspection of these materials can be achieved.

The focus on NDE is in the region of welds, as specified in the ASME Section XI Code requirements. However, as materials age, the base material may become more susceptible to degradation. In most power plants, the welds are only a very small volume percentage of components. Thus, even if the base material has a much lower failure rate, because there is so much of it, base material failures must be considered. There is a need for RES to investigate this potential problem.

In addition, the NRC has downsized by eliminating the NDE Mobile Laboratory that was previously located at Region 1. This loss, and the attrition/retirement of knowledgeable NRC NDE staff, is reducing the NRC expertise for reviewing problems when they arise, such as the recent pipe crack at V.C. Summers. The NRC needs capable staff as the industry moves towards imaging technology for conducting ISI. This technology requires special skills, based on knowledge and experience, to develop review capabilities.

<u>Imaging Science for Measurement of Weld Degradation in Light Water Reactor Systems</u>
The NDE Reliability research future includes work on the imaging science of welded components. This science will lead to the recognition of cause and effect relationships in ultrasonic data for prognostics, early detection of precursors to flaw initiation, and the assurance of high quality inspections.

Imaging significant phenomena depends on knowing what to look for and on careful examination for the unexpected. The detection of smooth planar flaws in images is an example of the latter. Imaging of repair cavities in components has been shown to be important, because recent work has shown that they have the highest likelihood to contain large flaws. Imaging of planar flaws that do not lie in a single plane is an example of important requirements for an advanced characterization system.

Performance metrics for imaging systems should be established. The complex baselines of responses in the pre-service state of welds should be available. Low observable degradation can be observed in contrast with such baselines, and imaging of microstructure is part of such techniques. Calibrated responses may play an important role in all images.

Advanced NDE imaging capability is becoming available. Improving the ability of NDE imaging systems to perform full characterization of the material should be a research objective. Improving resolution and using electronics for high dynamic range can be examined. New imaging hardware can be developed for low cost computing.

Also, NDE data should be improved to permit re-use. The relationship between the reproducibility of images and the dynamic range of ultrasound should be studied. Relational databases for evidence of cause and effect relationships in weld images should be developed. Web publishing of NDE data will improve image quality and the ability of the NRC or others to review and assess NDE inspection results. High value and low cost NDE will not be achieved until significant data re-use is made possible. The reliability of NDE images will be significantly improved when all of the significant degradations can be visualized. Advances in imaging systems, such as SAFT-UT hybrids with phased arrays, will make for high performance images. Degradation will not be routinely detected until the low amplitude observables are present in the images.

Research Involving Probabilistic Structural Mechanics Models

Since NDE and flaw assessment are intimately related, there are needs for improvements to the tools used in flaw assessments in order to maintain safety margins and reduce conservative assumptions for inputs to these calculations. The following areas of research are identified for supporting NRC decision making in the areas of risk-informed inservice inspection, nuclear power plant aging, and licensing of casks for shipping and storage of spent nuclear fuel.

Flaw Distributions for Welds in Piping and Spent Fuel Casks

To support pressurized thermal shock regulations, NRC has funded considerable research in collecting data and in developing methods for estimating the number and sizes of flaws in reactor pressure vessel welds. A similar need exists for welds in thinner sections, other materials, and other welding processes representative of a broader class of nuclear components. A short study was performed in 1996 that met the immediate needs of the Surry-1 pilot application of the risk-informed inservice inspection methodology. A follow-up study to refine the estimation procedure and to broaden the scope of work was never funded. Meanwhile the NRC staff continues to review industry submittals based on the ad-hoc evaluations performed in 1996. The following research needs efforts are identified:

- Review and update the piping weld model in the PRODIGAL computer code,
- Address additional materials and welding processes applicable to spent fuel applications,
- Perform parametric calculations for a range of materials and weld representative of piping and spent fuel applications, and
- Develop a set of correlations for estimating the number, sizes and locations of welding flaws.

Probabilistic Fracture Mechanics Codes

The probabilistic fracture mechanics code PRAISE was developed in 1981 to predict piping fatigue failure probabilities as part of the seismic safety margins project. Subsequent enhancements to the code have addressed stress corrosion cracking, inservice inspection, and fatigue crack initiation. Recent years have seen expanded applications of probabilistic methods as risk-informed approaches have become part of NRC's regulatory process. However, many key elements of the PRAISE code have never been updated and are no longer based on recent developments in both fracture mechanics methods and data on crack growth rates. Additional research is needed to address the following:

- Update the crack growth rate equations using the current data on fatigue crack growth rates that include the effects of environment and strain rates,
- Critical crack sizes based on elastic-plastic fracture mechanics, failure analysis diagrams, and stresses due to bending in addition to axial loadings,
- Piping failures due to wall thinning by the mechanism of flow assisted corrosion,
- Improved accuracy and computational efficiency by use of importance sampling and automatic control of stratified sampling,
- Simulation of the random nature of loadings and stress cycles, and
- Improved structure for data input and use of interactive menus.

High Burnup and Spent Fuel Issues

Issues Concerning High Burnup Fuel

This research recommendation includes the characterization and modeling of the rim restructuring phenomenon to determine its affect on the thermal properties of the fuel (needed for in-core operations and dry or wet storage) and the mechanical properties, which are of interest for off-normal or accident conditions. Some research on high burnup rim restructuring is being performed as part of DOE's NERI program, and it appears that the restructuring is more widespread in fuels than initially reported. Considering some of the results obtained for the Yucca Mountain Project, cladding integrity and failure mechanisms for spent fuel are not well understood. This is especially true for higher burnup fuels where the oxide layers are thicker and more hydrides are present.

Dry Storage Issues

In addition to the need to determine how high burnup fuel will behave under dry storage conditions, recent work has suggested that the present level of drying may not be adequate. The presence of water in a dry storage cask can result in a number of mechanisms that degrade the fuel or cladding. Possibly, an inexpensive and relatively easy process can be designed to further reduce the potential for these consequences.

There is also a need for NRC to verify the integrity of cladding in storage. Specifically, since pin-hole or hair-line cracks in the cladding are of major concern to the repository (because fuel interactions are quite rapid in a geologic time frame and thus these small defects can be as bad as gross defects- even worse if we consider that gross defects will be treated but small defects may be ignored), it will be necessary for NRC to verify whether present methods (sipping, ultrasonic testing, in-core water chemistry monitoring) are sufficient and conservatively bound the number of "failures." The drying process identified above would account for failed fuel and help ensure that water would be removed from these defected rods. This becomes more important for transportation, when temperatures will increase due to the poor heat transfer capabilities of the casks.

Repository Issues

Related to the Yucca Mountain Project (YMP), it may be important for the NRC to verify models that the Office of Civilian Radioactive Waste Management (OCRWM) is proposing. This will include verifying some of the experimental results on degradation mechanisms and kinetics of the fuel, cladding, and waste package materials. It will also include verification of the YMP FEP screening process where different Features, Events and Processes were considered and screened for inclusion or exclusion for consideration and modeling. NRC may need a basis for accepting or rejecting the recommendations of OCRWM on each of these FEPs. Paramount here is the fact that, in the testing that has been performed, irradiated materials have behaved markedly different than the unirradiated materials. This is true of the commercial spent fuel oxidation tests, irradiated cladding tests, N-reactor oxidation and drying tests, etc. It is important to stress that NRC testing on unirradiated or uncorroded materials may not be sufficient for adequate modeling. Hot cell work on real materials is essential.

Radiobioassay and Internal Dosimetry Concerns

NVLAP for Personal Air Sampling and Radiobioassay

The NRC should consider assessing the need for and impact of a National Voluntary Laboratory Accreditation Program (NVLAP) for personal air sampling and for radiobioassay measurements when those results are used to compute the dose of record. External dosimeters for personnel monitoring must pass NVLAP accreditation. Yet on legally equal status are doses computed from personal air samples and from radiobioassay measurements, neither of which are required to be from accredited programs for NRC licensees. Furthermore, NRC does not accredit internal dosimetry programs against current standards like those in the U.S. Department of Energy.

Investigation of Alternative Statistical Methods for Assessing Radiation Dose from Bioassay Methods

Bayesian statistical methods are well established in probabilistic risk analysis (PRA) for nuclear reactor accidents. There are some beginnings of use of Bayesian methods for bioassay and internal dosimetry within the U.S. DOE. The NRC should conduct a benchmarking performance assessment of Bayesian methods to reach consensus on their use in areas regulated by NRC.

Investigation of the Impact of Uncertain Dose Estimates in Regulation

The NRC should consider performing research in support of regulation on the basis of dose estimates that are uncertain, such as those that result from air sample or bioassay results, especially in the presence of respiratory protection. Current dose assessment techniques can give a point estimate of dose, e.g., 4.9 rems, but the reality is that answers are probability distributions, so that one could say "the median of the probability distribution of dose is 4.9 rems, the mean is 8.2 rems, the 5th %ile is 0.84 rems, and the 95th %ile is 28.6 rems." Should NRC regulate on the median, the mean, or some specified percentile in order to ensure adequate protection of the worker and the environment? Some work has been done on this by NIOSH in the 1970s, but it is not in use.

Research Concerning Risk Assessment for Radioactive Contamination

Quantitative Historical Consequences of Radiological Accidents and Events Involving Contamination or Release of Radioactive Material

The NRC should consider conducting research to determine quantitatively what consequences have occurred from accidents and incidents, and even from routine operations, involving byproduct materials. NRC- NMSS has performed preliminary research for sealed sources, but has done little that was quantitative along these lines for routine contamination or accidents involving byproduct materials. The existing work should be extended to include a complete study of the consequences of the orphan source phenomenon.

Discounting Future Risk for Risk Assessment of Radioactive Waste Disposal and Decommissioning

Current practice for risk assessments for radioactive waste disposal includes risk assessments out to 10,000 years. This time frame is unparalleled in any other human endeavor, especially in areas of chemical disposal and land use such as strip mining. Some compounds have "half-lives" in particular contexts; elements and other chemicals have infinite half-lives. The NRC should consider commissioning a study to examine the use of discounting across all risk management, in particular, for radioactive waste disposal.

Improving Regulations for Currently Operating Nuclear Facilities

Cyber Security

Within the year, the NRC will promulgate proposed rulemaking regarding cyber security at nuclear generating stations. To date, no investigation has been conducted to address the vulnerability of a nuclear facility to cyber intrusion. The Department of Energy Infrastructure Assurance Outreach Program is commencing a small program to address this issue. The NRC has chosen to not participate in funding this effort, beyond establishing an NRC interface. The program is under-funded, and certain items the NRC desires to have addressed may not be accomplished due to lack of funding. Participating in funding this program could assure that the proposed rulemaking would be predicated on a sound regulatory foundation.

Pressing Research for Currently Operating Nuclear Facilities

Deviation from the Analyzed End-of-Cycle Flux Shape for BWRs

Maintaining the proper flux shape at the end of an operating cycle at a BWR is crucial for both operation and to maintain the safety envelope. However, the establishment of an appropriate critical power ratio penalty (Delta CPR) to be taken if the flux shape becomes too top peaked has never been addressed.

At the end-of-cycle conditions in a BWR, all control rods are fully removed from the core. The cycle thermal limits are calculated assuming a "Haling" flux shape at end-of-cycle conditions. If the flux shape is more top peaked than the assumed shape, the control rods require a longer insertion time to reach the "meat" of the flux and suppress it during a transient. This difference can be quite significant. Benchmark calculations should be performed with various flux shapes to obtain some sensitivity regarding how impacted the end-of-cycle scram is to top peaked fluxes.

Of interest, a change in shape is accounted for in a final feedwater temperature reduction analysis, but has never been addressed in a cycle analysis. This is a known deficiency in the BWR safety analysis.

Conscientious Station Nuclear Engineers will assiduously maintain flux shapes, and if the shape becomes too top peaked, will partially insert a rod or two per quadrant to assure that a scram engendered from a transient will have an adequate "purchase" or "bite" to suppress the flux. This action is not required by any Technical Specification.

Boiling Water Reactor Feedwater Flow Venturi Calibration

Accurate calibration of feedwater flow venturis are required to assure that the reactor is within the bounds of it's licensing analysis. Almost all of the calculated heat balance value is based on feedwater flow rate and feedwater temperature into the reactor vessel. The power range nuclear instrumentation is calibrated to this heat balance. Core power and fuel thermal limits are predicated on the heat balance value. It is a critical parameter.

Operating BWRs must calibrate their feedwater flow venturis in-situ, due to their being contaminated. National calibration laboratories (e.g., Wiley Labortories) will only accept new venturis for calibration.

Currently, venturi calibration at BWRs is performed by injecting a compound containing a radioactive alkaline metal into the feedwater stream, and counting the activity downstream of the feedwater flow venturis. The alkaline metal is highly soluble, powerfully electrovalent, and will plate out on the piping between the injection point and the detection point - causing inaccuracies in the calibration process.

Several ideas are available concerning better calibration methods, but funding for these initiatives has not been available. The NRC should consider at least partial funding for such an initiative.

Future Directions for NRC Research
Sandia National Laboratories

March 15, 2001

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The subjects contained in this brief paper have been identified by senior staff and management at Sandia National Laboratories (SNL) as ones that should be part of the NRC's forward looking research program. This is neither a complete nor prioritized list, rather it represents insights from SNL's experience base that suggest major research topics that would help NRC to continue successful execution of its mission.

RESEARCH SUPPORTING REGULATORY PROCESSES FOR NEW REACTOR TECHNOLOGIES & DESIGNS

Current electricity supply and price issues, along with traditional fossil fuel plant emissions and global warming concerns, have created a strong potential for a new nuclear power era. Advanced plant designs may now become a reality within the next ten years, rather than the previously thought 30 to 50 years. The single greatest contribution the NRC can make at this time to the future of safe, efficient, environmentally friendly nuclear power regulation is to initiate a research program to support effective resolution of anticipated licensing issues associated with new reactor concepts. The NRC should work with industry, universities, national laboratories, and DOE/NE to identify the range of candidate designs being considered, then develop and implement a research program based on the common regulatory issues these new designs will present. The technology outlines of new plant designs (variously noted as "generation 3 1/2 or IV") are becoming reasonably well bounded by current industry work and government sponsored projects associated with NERAC, NERI, I-NERI, and NETP. Thus, it should be possible to identify the likely new technologies that may impact risk informed licensing of new plants, behavior of advanced fuels, and environmental considerations relative to irradiated fuel management and health effect standards. These in turn will point the way to the triad of modeling, experimental, and analyses activities necessary for the NRC to be in a position to conduct prompt, effective licensing of new plants.

RISK INFORMED REGULATION

After nearly 30 years of PRA research and application, the NRC is moving aggressively to incorporate risk insights into the regulatory process. While past research has been highly successful, much remains to be done before the full benefits of risk informed regulation can be achieved. It is extremely important that this work goes forward, as it perhaps represents the area of highest payoff in terms of safety benefits and reduced cost to both the NRC and the industry. However, if risk informed regulation goes forward based on incomplete and poor quality PRAs, the result is likely to be decisions that are difficult to defend and the potential loss of public confidence. Major improvement should include:

- Current Facilities -- Implementation of risk informed processes is made more difficult due to a lack of complete and accurate risk information. Current IPE and IPEEE information is incomplete and varies in quality. One important research area is the development of a complete set of PRA standards along with agency guidance for their use. But before such a set of standards can be finalized, there are a number of PRA topics that need further attention, including digital I&C, human reliability, plant aging, fire, and low power/shutdown risks. Along with the treatment of these issues, the ultimate use of PRA in decision making is still problematic. Additional work on the treatment of uncertainties and their accommodation in a robust decision process incorporating traditional defense-in-depth and conservatism concepts is appropriate. Currently, decisions are often made based on point estimates with little understanding of the uncertainty ranges. As part of the support for PRA implementation, the Office of Research should support the development of additional staff training in PRA and Reactor Safety in general.

- Future Facilities -- A completely new regulatory framework is needed for plants that are different in nature from current LWRs. A top-down framework is needed that builds off of the current Safety Goal Policy and develops functional requirements for each new reactor type. A generic approach should be developed that can then be adapted to each new design. Early cooperative activities with the Department of Energy could effectively leverage resources in this area and help focus DOE research as well. Along with a general framework, generic technology issues that can be addressed include software reliability and the treatment of smart equipment. Risk assessments to confirm portions of the applicants' safety cases will be needed. As each reactor technology is brought to the NRC, a research plan for collecting needed risk information should be developed. This plan will guide technical input from other parts of the Office of Research to support the risk-informed decisions. Significant training in advanced PRA methods and the safety of advanced designs will be needed for the NRC staff. The Office of Research should support this training.

FIRE PROTECTION RESEARCH

Fire protection for nuclear power plants is currently based on prescriptive rules. Despite this, fire risk assessments performed a part of the IPEEE effort indicate that the frequency of fire-induced core damage accidents is comparable to, or in some cases even higher than, the core damage frequency associated with all other initiators during normal plant operation. This suggests that the continuation and enhancement of NCR's fire research is essential. Major enhancements should include:

- Ability to independently analyze fire scenarios using up-to-date fire modeling tools -- An improved model would allow NRC to make more realistic, open, and technically defensible judgements of the fire threats posed to systems and fire protection barriers in nuclear power plants. Such a model would treat the degradation of instrumentation, control, and power circuits, including associated cabling, plus operating personnel performance, due to heat, smoke, and gasses. A mechanistic capability to model fires together with existing probabilistic capabilities would allow risk-informed evaluations of designs and alterations to fire protection systems proposed by licensees to be more effectively evaluated.

- Development of risk-informed fire protection regulations -- Recent efforts supported by the NRC have led to the adoption of a National Fire Protection Association standard (NFPA-805) on performance-based fire protection for nuclear power plants. However, early indications are that this standard does not meet industry's needs and will not be adopted voluntarily. Hence, the challenge of developing a framework for implementing risk-informed fire protection that is acceptable to both industry and the NRC remains. Coupled to this goal is the general topic of fire risk analysis methods, expectations and standards. Currently, there is no concise documentation of methods of fire risk analysis that would meet NRC staff expectations with regard to risk-informed regulatory decision making. Development of such guidance would be prudent.

CONTAINMENT INTEGRITY RESEARCH

As the next generation of NPPs are designed and licensed, the NRC's research investment in understanding the response of containment structures to accidents beyond their design basis should provide significant regulatory insights. Past experience in instrumentation and testing large-scale containment models can be extended to:

- Development of recommended capacity criteria and methodologies to analytically derive containment capacities, including addressing of uncertainties.
- Identification of requirements for integration of instrumentation into future NPP containment structures to enable continuous monitoring of containment integrity. Accident management could be improved by real-time feedback on containment response to plant managers, and for post-accident/earthquake assessment of containment integrity. An 'open' monitoring environment would be possible that may be attractive to regulators and the public.
- Development of containment/confinement requirements for 'inherently-safe' NPPs. Some next generation NPPs are being proposed as 'inherently safe' with respect to severe accidents, however, some degree of containment or confinement may still be desirable from the viewpoint of public acceptance and/or required by external loads (seismic, wind, etc.) and physical security protection.

SEVERE ACCIDENT RESEARCH

Over the last twenty-plus years the NRC has conducted a massive study of the phenomena associated with severe reactor accidents, and has set about consolidating the information in the MELCOR computer code for the analysis of severe reactor accident progression. There are, however, some technical issues that have not been fully resolved by past research or have emerged as a result of changes in the nuclear power industry. Certainly, the possibility of air ingression into the reactor vessel and interaction with residual fuel following reactor vessel rupture is an open technical issue that the staff is investigating in its collaborations with the various parties carrying out the PHEBUS-FP experiments. Another issue is the deposition of aerosol particles on surfaces within the secondary side of steam generators during accidents involving steam generator tube rupture. This issue is being studied in the collaborative program called ARTIST. In addition, in view of current plant life extension plans for existing reactors, modeling enhancements with respect to higher burnup, mixed oxide, and on-site fuel storage in both pools and dry casks should be addressed. Capabilities to analyze fuel pool accidents, and safety margins with storage of high burnup fuel in dry casks should be considered. Other issues that deserve continued attention are:

- High Burnup Fuel Foaming & Degradation -- Past core degradation testing and analysis has been with fuels of modest burnups relative to those becoming common in operating plants today, and certainly of low burnups relative to those envisioned in next generation reactor concepts. Fuel degradation and associated alteration of coolant flows under accident conditions by foaming rather that the usual melting and candling mechanisms should be added to NRC's suite of modeling tools.

- High Burnup Fuel Fission Product Releases -- Similar to fuel foaming, models of fission product releases have been devised based on test results for fuels taken to burnups that are substantially less than burnups that are now becoming common or anticipated in advanced designs. These models may not adequately account for the substantial changes in morphology and chemistry that take place as fuel is taken to higher burnup. There is a need, then, to revise these models to account for the increased release rates that are possible from fuel with the high burnup microstructure.

- Effects of Electrostatic Charging on the Behavior of Nuclear Aerosols -- Current models of the growth and deposition of radioactive aerosols produced during reactor accidents consider diffusion, turbulent processes, phoretic processes, and gravitational settling. But they do not consider effects of the electrostatic charging one would expect to occur due to the difference in mobilities of positive and negative ions in an irradiated atmosphere. Charge buildup could affect the rates at which deposition processes remove fission products from the gas phase and attenuate the consequence of a reactor accident.

- Anticipation of New Plant Designs -- In the area of anticipatory research, NRC Research codes will be called upon to evaluate future reactor designs. These will range from evolutionary designs such as AP1000 to quite revolutionary designs such as the Westinghouse IRIS design, General Atomics modular high temperature reactor, and Eskom's pebble bed reactor. The latter two reactor designs require modest to significant extensions to current capabilities. It will be necessary to begin code enhancements sufficiently in advance of requests for licensing in order to provide timely assessment of source term issues, containment design, fuel performance and plant siting.