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LICENSING ISSUES ASSOCIATED WITH THE FUTURE GENERATION OF NUCLEAR POWER REACTORS

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PRESENTED AT THE OPEN PLENARY SESSION OF THE
FIFTH INTERNATIONAL TOPICAL MEETING
ON NUCLEAR REACTOR THERMAL HYDRAULICS,
"TOWARDS THE NEXT GENERATION
OF NUCLEAR POWER REACTORS"
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Good morning, ladies and gentlemen. It is indeed a pleasure to be invited to participate in this international topical meeting on thermal-hydraulics for the next generation of nuclear power reactors. I'm pleased to see the large attendance, in particular the large number of international participants. Several days ago I returned from St. Petersburg, Russia, where I participated in the third annual meeting of the Nuclear Society of the former Soviet Union, in which there was a large number of international participants. Without question, nuclear technology is an international technology.

I am very pleased that technical conferences such as this one are being sponsored by the American Nuclear Society. I consider such topical meetings vital to the advancement of science, safety, and performance of our present and future nuclear facilities. This conference is important as the world embarks on designing safer and more reliable nuclear power plants. The European Community is engaged in designing advanced evolutionary reactor plants, Russia is embarking on new reactor designs, China is reportedly expanding its nuclear industry, Japan and Korea are continuing to expand their reliance on
nuclear energy, Taiwan is expected to place an order for two nuclear units in 1993, Indonesia is embarking on a nuclear power program, and the United States has taken the lead by designing several power reactors of the next generation.

While my discussions will focus on the longer term activities related to advanced reactor designs, I believe it important to note that the NRC staff expects to issue within the next few months the final safety evaluation report on General Electric's Advanced Boiling Water Reactor (ABWR), and in 1993 the final safety evaluation report on ABB-Combustion Engineering's System 80+. These nuclear plants are known as evolutionary advanced light water reactors (ALWR). They incorporate the lessons learned from over 1400 reactor-years of U.S. operating experience and extensive research conducted during the past three decades. The NRC is also reviewing several advanced light water reactor designs which employ passive safety systems and is reviewing the preliminary designs of advanced heavy water, liquid metal, and gas-cooled reactor plants.

Deployment of these advanced nuclear power plants will depend on the public's acceptance of the improved nuclear technology. To gain public acceptance, present reactors must continue to operate safely and reliably, and the next generation of reactors must be accepted as more tolerant of malfunctions and human errors.

To develop safer reactor designs and gain public acceptance of those designs, both the nuclear industry and the safety regulators must demonstrate a level of technical expertise which results in public and investor confidence. Thermal-hydraulic analyses and experiments will play a vital role to that end.

During the past 30 years, the industry and the academic community have significantly advanced their analytic capabilities to model transient and accident events on the computer. They have advanced from the original homogeneous-equilibrium, fixed 3-node computer codes, to the variably noded computer program, known as FLASH, developed by John Redmond, while working at the Bettis Laboratory. I would characterize the FLASH code as the father of today's variably-noded, multi-dimensional, and non-homogeneous/non-equilibrium computer programs. We have evolved from executing these computer programs on a large, but slow mainframe, to executing them on significantly faster and more powerful desk-top work stations.

We initially focused our analytic modeling capabilities on large and small break loss of coolant accidents. Following the Three Mile Island accident, we centered our efforts on transients and accidents with multi-component failures and refined our analytic capabilities to evaluate small break accidents with greater accuracy. We improved and are continuing to improve our analytic capabilities to assess the consequences of postulated severe core accidents, from time of event initiation to containment failure -- including the offsite consequences of those postulated events.

In the late 1980's, the NRC concluded that, for the present generation of reactor designs, we had reached a state of analytic capability adequate for auditing Appendix K (of 10 CFR Part 50) licensing submittals and for independently assessing operating reactor events and safety issues. As a result of that decision, the NRC reduced its level of effort in the area of thermal-hydraulic model development to what is termed a "maintenance" level of effort. For severe accident analyses, the NRC continued to support code development activities and entered into international cooperative programs on severe accident phenomenology.

In the late 1980s, the nuclear power plant vendors in the United States began knocking at the NRC's door requesting reviews of their new nuclear plant designs. The industry, through the Nuclear Power Oversight Committee (NPOC), issued its Strategic Plan for Building New Nuclear Power Plants, the Department of Energy (DOE) issued its National Energy Strategy, and the Electric Power Research Institute (EPRI) requested the NRC to review the Utility Requirements Document for the Advanced Light Water Reactors. Through these industry and government activities, the goal was established to have advanced nuclear power plants constructed and operating around the year 2000.

However, these industry and government initiatives caught the NRC off-guard. While the NRC had a modest effort ongoing to review General Electric's ABWR and ABB-CE's System 80+ designs, the NRC found itself in a catch-up mode to respond to the industry's increased level of activity. It was quickly recognized that many of the knowledgeable technical reviewers who were needed to respond to the industry's initiatives were either scattered throughout the agency -- performing other activities -- or, in some cases, were no longer employed within the agency.

Recognizing that the agency was not carrying out its responsibilities to review the new designs in an optimum and timely manner, the Commission directed the staff to place greater emphasis on the NRC's licensing review capabilities. Under the competent leadership of the EDO, Mr. Taylor, and the Director for the Office of Nuclear Reactor Regulation (NRR), Dr. Murley, the NRC rapidly met the emerging challenges. In addition, the lead management responsibilities for all of the advanced reactor designs were centered within NRR, the office with licensing responsibility.

The NRC is committed to conduct thorough reviews of the industry's submittals. I emphasize thorough reviews, because the new designs are being reviewed under the new and more stringent certification requirements of the NRC's "... Standard Design Certification Rule ...," 10 CFR Part 52. This rule requires that all safety issues be resolved prior to the certification of any nuclear power plant design.

As the NRC staff engaged in the reviews of the pre-application submittals for the various advanced plant designs, some criticisms began to surface over the effectiveness of the maintenance programs in sustaining technical expertise in the thermal-hydraulics disciplines.

While there appears to be general confidence in the thermal-hydraulic computer codes, numerous concerns have been raised regarding the current condition of the codes (They are a patchwork of past fixes and are not fully up to date with respect to physical correlations, models, and computer hardware and software); ease of use and adaptability of the codes to modern workstations, such that the codes would be used more extensively; the sufficiency of current maintenance programs to fix known problems; and, the sustenance of the expertise available to the NRC to address problems in such areas as modeling, code development, validation, and the planning for the conduct of tests.

Perhaps, if there is a lesson to be learned from our past code "maintenance" practices, it is that maintenance should not be restricted just to maintaining the code operational on a computer. I believe that maintenance must include preserving the technical capabilities needed to modify and improve many of our extremely complex computer codes. Further, I believe that means that maintenance must include succession planning for the next generation of model and code developers, and it must ensure intellectually challenging assignments if we are to keep the technical experts motivated and available when safety concerns arise. It must also include adequate resources and priorities for correcting code deficiencies and errors. In the United States, the NRC and the Department of Energy must coordinate their efforts to ensure that this national resource of expertise, facilities and codes is sustained at levels sufficient to support future national needs.

The matter of maintaining technical capabilities in the face of limited resources and program growth is not limited to the thermal-hydraulic areas, but also applies to the material sciences, reactor physics, and instrumentation and control disciplines. The underlying objective is to maintain technical competence readily available to respond to the challenges of the future. This objective is being pursued by countries such as France, Germany, Indonesia, Italy, Japan, Russia, Sweden, Taiwan, and the United States.

The NRC's Advisory Committee on Reactor Safeguards, its consultants, and others have been concerned over this issue, and the Commission is giving increased attention to this matter. Just recently, the Commission endorsed a recommendation made by its Advisory Committee on Reactor Safeguards to establish a review group of experts to assist the NRC staff assess the status and needs of the NRC's thermal-hydraulic capabilities. In a few moments I will indicate some of the thermal-hydraulic policy issues that the NRC staff has identified for careful review.

In the area of advanced reactor design reviews, you may be aware that the NRC has received an application from Westinghouse for the design certification of its Advanced Passive 600 Mwe nuclear power plant, known as the AP600. The NRC has also recently received an application from General Electric for the design certification of its Simplified Boiling Water Reactor, known as the SBWR.

In parallel, the NRC is reviewing preapplication submittals of General Electric's Power Reactor Innovative Small Module (PRISM), also referred to as the Advanced Liquid Metal Reactor (ALMR), AECL Technology's Canadian Deuterium Natural Uranium Model 3 (CANDU 3) reactor, General Atomic's Modular High Temperature Gas-Cooled Reactor (MHTGR), and the Asea Brown Boveri-Combustion Engineering (ABB-CE) Process Inherent Ultimate Safety (PIUS) reactor designs.

In 1991, the NRC staff established its schedules for issuing the preapplication safety evaluation reports for these advanced reactor designs. The original schedule had the staff issue preapplication safety evaluation reports for the ALMR and MHTGR by the end of calendar year 1992, and safety evaluation reports for CANDU 3 and PIUS by mid-calendar year 1993. However, it became apparent that the NRC could not devote the resources necessary for parallel preapplication reviews. Therefore, the staff has been meeting with the reactor vendors to discuss the vendors' updated needs and schedules.

Based on the more recent discussions held between the NRC staff and the applicants, the projected dates for the formal submittals for these advanced reactor designs appear to be as follows:

1. For the ALMR, the application for a preliminary design approval (PDA) is scheduled for calendar year 1995. The formal application for design certification is projected between the years 2000 and 2003.

2. For the MHTGR, the dates for application for a PDA and formal application for design certification will be established in 1993.

3. The application for design certification of the CANDU 3 power plant is scheduled for calendar year 1995 or 1996.

4. The application for design certification of the PIUS reactor design is scheduled for calendar year 1994 or early 1995.

Each of the proposed reactor designs will pose new technical challenges and policy issues for the NRC. The review approaches for these designs have not been fully developed. Where the NRC staff identifies differences between the existing LWR criteria and the criteria which should be applicable to the design in review, or where the pre-applicant has justified a less conservative approach than that prescribed by the existing criteria, the new criteria will be submitted to the Commission for approval. Ultimately, such criteria will be codified through the normal rulemaking process or by the design certification rulemaking process for that specific design.

Some examples of potential policy issues for the advanced plant designs undergoing preapplication reviews include:

1. Identification of appropriate transient and accident event categories and evaluation criteria.

2. Development of mechanistic source terms.

3. Alternative approaches for satisfying the requirements of traditional containment structures.

4. Consideration of modifying the requirements for the emergency planning zone.

5. Consideration of a single reactivity control system without diversity (i.e., PIUS design).

6. Consideration of reducing the staffing required to operate the reactors.

7. Criteria for identifying the safety-related structures, systems, and components.

8. Consideration of reliance on a completely passive system as the single safety-grade residual heat removal system.

9. Consideration of the acceptability of a design in which the reactivity tends to increase under specific conditions.

More than likely, the list will be expanded as the staff continues to review and evaluate the proposed designs. The resolution of these issues will heavily rely on the technical merits of the proposed justification presented by the industry and when necessary, independently confirmed by the NRC.

I stress how important it is for the vendors to provide thorough technical justification for their proposed designs and positions. This requires significant up-front financial commitment and continuous interaction with the NRC staff. The thermal-hydraulic community will play a vital role in the resolution of some of these issues. The design certification review process will be conducted more efficiently if the majority of the general criteria for the advanced reactors are established prior to the NRC receiving a formal application. This would require early commitment on the part of the applicant and the NRC to develop such criteria.

I would now like to highlight some specific thermal-hydraulic issues which have been identified by the NRC staff for the advanced reactor designs.

Advanced light water reactor designs, such as the AP600 and the SBWR, incorporate passive, gravity-fed and stored-energy-fed water supplies for emergency core cooling, and they use natural circulation for decay heat removal. The low-flow regimes and flow paths associated with natural circulation cooling in these designs are not typical of current light water reactors, and therefore the computer codes used for current LWR analyses need to be validated to reflect these aspects of these advanced reactors.

Presently licensed reactors and the advanced evolutionary designs use active components, such as pumps, to inject coolant into the primary reactor system. In the safety analyses for these plants, the amount of emergency coolant injected into the primary reactor system was known and conservatively modeled as a function of reactor coolant system pressure.

This is not the case for plants with passive emergency core cooling systems, where effects such as mass distribution, temperature differentials, condensation rates, stored heat energy, non-condensible gases and pipe geometries enter much more significantly into the calculation of the amount of emergency coolant that penetrates the primary reactor system. These effects place greater importance on the computer code's abilities to accurately model flow rates at low differential pressures. They place greater importance on accurately modeling the coolant inventory distribution, the different flow regimes, the stratified flow conditions, the stored energy and the thermal-hydraulic conditions within the steam generators and the containment building. Consequently, one must scrutinize in greater detail the accuracy of our computer programs and obtain an understanding of how these analyses can be performed accurately and conservatively.

The effects of non-condensibles, boron propagation, and reactivity feedback will likely play a greater role in these designs than in presently licensed reactor designs. These issues are being reviewed by both the reactor vendors and the NRC staff, and experiments are being identified for validating the computer programs.

For the advanced liquid metal reactor (ALMR), the thermal-hydraulic issues identified thus far include absorber and fuel/blanket flotation, flow blockage, and adequacy of natural circulation cooling. The applicant has stated that natural circulation will be verified in the ALMR prototype reactor.

There are a number of thermal and fluid-flow design issues which have been identified with the MHTGR. These include core flow distribution, flow oscillations, fuel-block displacement, and flow blockage. Another issue is the laminar flow that exists in the MHTGR at low power, low flow conditions, which could lead to flow stagnation and reversal.

Thermal-hydraulic modeling of the CANDU 3 design will require analysis of horizontal fuel geometries in pipes under stratified two-phase flow conditions. Before efforts to develop an input deck for the CANDU 3 power plant begin, a number of other tasks must be completed to establish an accurate thermal-hydraulic modeling capability. These tasks include reviewing the Canadians' thermal-hydraulic computer models for CANDU 3 and CANDU 6, and determining existing capabilities in the United Sates for modeling horizontal flow and other CANDU flow phenomena.

Another issue is the adequacy of using LWR heat transfer and flow correlations for CANDU, which uses heavy water. The extent to which these correlations have been benchmarked against heavy water experiments needs to be evaluated, and newer heat transfer and fluid flow correlations developed by AECL also need to be benchmarked.
Also, given the possible positive reactivity feedback which could accompany a CANDU 3 design under accident conditions, an appropriate neutronics model is needed.

The PIUS reactor poses additional challenges to the thermal-hydraulic computer programs. The PIUS design uses a large pool of borated water hydraulically connected to the primary coolant path for shutdown. The pool is connected at two points to the primary coolant system. The connection is made through a component called a density lock.

The operational performance of the PIUS reactor depends on the establishment and maintenance of a stable set of density locks during normal and transient operation. A basis for quantitatively predicting the behavior of the density locks must be developed. The NRC is presently sponsoring contracts which will investigate the hydraulic behavior of the density locks.

Another consideration for the PIUS design is low coolant flow velocities. Core flow velocities are particularly low during a scram, when the pumps are tripped and natural circulation is the driving force. Coolant velocities low as compared to the velocities in standard LWRs will require that analytical models, heat transfer correlations, and flow velocities be reevaluated for applicability to the PIUS analyses. The range of flow velocities for the correlations used in the present LWR codes may not be appropriate for the PIUS design. Studies initiated by the NRC will evaluate low core flow velocity heat transfer and flow parameters.

Review of the PIUS design will also include studies of the effects on core flow of the gas baffle which is located above the lower density lock. There is concern that the gas baffle may adversely affect flow to the core assemblies and the corresponding boron concentration, which might result in undesirable heat generation characteristics and reactivity variations within the fuel assemblies.

Another component of the PIUS design is the siphon breaker. A venturi-type siphon breaker is provided to assure that during a postulated cold-leg break the core is not uncovered. Performance of the siphon breaker in transient flow scenarios needs to be studied. The NRC has contractors evaluating the performance of the siphon breakers.

Other issues associated with the PIUS design include cooling of the pool water using natural circulation cooling, containment cooling, pressure suppression in the containment, once-through steam generator performance, and refinement of the TRAC computer code for application to PIUS.

The quality assurance, and validation and verification of the thermal-hydraulic and core nuclear codes used in the safety analyses for all these designs will be scrutinized very closely by the NRC.

In conclusion, the nuclear industry is actively pursuing innovative reactor designs to meet the future energy needs of this country. The U.S. industry has received support from the President of the United States, the Congress and the Department of Energy to design and certify safer and more reliable sources of nuclear-generated electricity for meeting future energy needs in the United States.

For you thermal-hydraulic specialists, we are in the early stages of exciting times. Thermal-hydraulic research and computer code development and refinement will play a very important role in the design, review, certification and operation of the next generation of nuclear power plants.

We must remain ever mindful of the technical expertise required to design, construct, instrument and operate the experimental facilities required for code development and validation. We must remain mindful also of the expertise required to develop and modify our computer programs to better assist the regulators and licensees assess operating reactor events, develop better operator procedures, and improve the fidelity and capabilities of the reactor simulators, to mention but a few examples.

Succession planning in all of the technical disciplines is vital if we are to be prepared to meet our future needs in a timely and effective manner. The nuclear industry is investing hundreds of millions of dollars preparing for our future energy needs. The NRC will meet its responsibility and duty of providing timely and efficient responses to the industry's applications and will meet its responsibility to the general public whose safety is of paramount importance and must be adequately protected.

I am pleased with the initiatives taken by the ANS to highlight and focus our attention on these activities by sponsoring this topical meeting, and I encourage broadened participation in these areas. The ANS can and does play an important role, through its members, in identifying public concerns and the resolutions to those concerns.

Thank you! I would be pleased to answer any questions you might have.