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TMI-2 Vessel Investigation Project

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Background

As the last of the fuel was being removed from the Three Mile Island Unit 2 (TMI-2) reactor pressure vessel, it could be seen that quantities of melted fuel had covered the lower head of the pressure vessel, and there was concern that the lower head of the reactor pressure vessel might have undergone chemical and thermal attack. The possibility of such an attack raised important safety issues relating to reactor vessel integrity following a severe accident.

In October 1988, the NRC, in cooperation with 10 foreign countries under the auspices of the Organization for Economic Cooperation and Development's (OECD) Nuclear Energy Agency, began a joint research program to examine and analyze material samples from the lower head of the TMI-2 reactor pressure vessel. The objectives of this program, called the TMI-2 Vessel Investigation Project (VIP), were to (1) investigate the condition and properties of materials extracted from the lower head of the TMI-2 reactor pressure vessel, (2) determine the extent of damage to the lower head by chemical and thermal attack, and (3) determine the margin of structural integrity that remained in the pressure vessel.

Prior to the initiation of the VIP, the Department of Energy (DOE) had supported extensive postaccident examinations and analyses of the TMI-2 damaged core. The primary objective of this TMI-2 Accident Evaluation Program was to develop an understanding of (1) core damage progression in the upper core region, (2) heatup of the consolidated region leading to extensive melting of the core, (3) relocation of approximately 20 tons of debris to the lower head, and (4) release of fission products to the reactor vessel and the containment.

DOE decided to end the DOE-sponsored research on the evolution of the TMI-2 accident following removal of the damaged core. In October 1987, a meeting was held at MIT to review the results and conclusions of the work completed up to that time. The review made it clear that core degradation and melting had posed a threat to vessel integrity. The review group, which included Professor Neil Todreas of MIT and Edwin Kintner of GPU Nuclear, recommended that the issues relating to vessel integrity should be investigated further. RES agreed and decided to perform additional research, subject to obtaining the cooperation of GPU Nuclear, for the purpose of determining what the margin to reactor vessel failure had been during the course of the accident.

The principal conclusions from the DOE-sponsored research were that the TMI-2 core damage progression involved the formation of a large consolidated mass of core material surrounded by supporting crusts, the failure of the supporting crusts, and finally, the long-term

cooling of a large volume of molten core material. The TMI-2 accident demonstrated that a severe accident can be terminated and confined to the reactor pressure vessel by cooling water before the failure of the lower head. However, there was no quantitative information that could be used to determine how close the vessel was to failure.

The VIP examinations described in this article go beyond the work performed under the previous TMI-2 examinations. Specifically, the VIP plan was to obtain and examine samples of the lower head steel, instrument penetrations, and previously molten debris that was attached to the lower head and use this information to estimate the vessel margin-to-failure. These tasks were not included in the scope of the DOE Accident Evaluation Program.

VIP Scope of Work

The scope of work of the VIP consists of three major components—sample extraction, sample examination, and analysis of results. The sample extraction phase involved developing and testing an extraction tool and procedures, extracting the samples from the lower head, and transferring the samples from the TMI-2 site for examination. The criteria and constraints that were considered in developing the extraction tool included:

- Obtaining the largest number of samples possible during the 30-day window that was available for the extraction process,
- Not breaching or significantly weakening the reactor vessel, and
- Working on a shielded platform mounted 40 feet above the lower head and remotely extracting the samples that were covered by highly borated and buffered water.

The cutting technique that was selected for extracting samples of the lower head steel was a metal disintegration machining (MDM) process. This process uses a series of electric arcs to melt a small amount of material in the cutting area. Water pumped through holes in the cutting electrode cools the molten material and flushes the resulting particles away from the cutting area.

Although only a limited number of samples could be extracted, different regions of the lower head had to be sampled. The following areas were selected for extracting samples.

- As close as possible to the area directly beneath the primary relocation path of molten core material to the lower head,
- Toward the radial center of the lower head underneath the maximum debris thickness,
- In the quadrant of the lower head where a "wall" of consolidated debris similar to a lava front had been observed,
- In an area of the lower head not contacted by the molten core material to act as a "control" sample, and
- Areas that include one or more instrument penetrations, especially in the areas noted above.

Since this was a first-of-a-kind process using a specially designed MDM cutting head, the exact number of samples could not be predicted in advance. It was hoped that from 8 to 20 samples could be obtained. As it turned out, the sample extraction was very successful, and 15 reactor vessel steel specimens, 14 incore nozzles, and 2 incore guide tubes were extracted from the lower head over a 30-day period ending March 1, 1990.

Following extraction, the vessel steel samples were decontaminated, sectioned, and distributed to Argonne National Laboratory (ANL), Idaho National Engineering Laboratory (INEL), and seven participating countries for mechanical and metallographic examinations. Also, the nozzles and guide tubes were cut and distributed for examination to ANL, INEL, and the CEA in Saclay, France.

Since the extraction of the test specimens in 1990, metallographic examinations of the vessel steel

samples, including microstructural examinations and hardness measurements, have been completed to determine the temperature distribution in the lower head that existed during the accident. Testing of the mechanical properties of the samples was also performed to provide data on ultimate strength and change in resistance to failure by creep rupture for calculating the margin to failure.

The VIP team removed more samples from the damaged reactor vessel than they originally expected, and the preliminary examinations indicated that comprehensive studies of these samples would provide important data on the behavior of reactor vessels during severe accidents. As a result, in September 1991 the VIP Management Board decided to amend the original agreement to extend the VIP program from September 30, 1991, to March 31, 1993, and to increase the budget by approximately $1.5 million. The objectives of the amended program are to (1) perform more detailed testing and examination of the in-core instrument tube nozzle penetrations and the in-core instrument guide tubes that were extracted from the lower head, (2) perform additional analyses of potential reactor vessel failure modes based on data from sample examinations, and (3) assess the margin-to-failure of the lower head of the reactor vessel.

Metallographic and scanning electron microscope (SEM) examinations of the instrument tube nozzles have been performed to estimate the nozzles' temperature and interaction of the Inconel 600 nozzle material with molten core materials. Results of these examinations have been used in conjunction with estimated temperature distributions to assess potential failure modes of the instrument tube penetrations.

The Accident Evaluation Branch and Materials Engineering Branch of RES are jointly funding approximately 50 percent of the cost of this $9 million program. The remaining funds come from the OECD member countries participating in this project (Belgium, Germany, Finland, France, Italy, Japan, Spain, Sweden, Switzerland, and the United Kingdom) and the Electric Power Research Institute.

Future Plans

Results of the ongoing TMI-2 lower head examinations are expected to provide additional information on the physical properties of the specimens, temperature distributions in the instrument nozzles, and interactions between the molten core material and the vessel. These results will be used to perform scoping analyses of potential reactor vessel failure modes, such as global or local failure of the reactor vessel lower head and penetration tube failures (i.e., tube heat-up and failure resulting from the flow of molten core material in the instrument tube, as well as tube ejection following heat-up and failure of the weld on the inner surface of the reactor pressure vessel). More detailed analyses of the most likely failure mechanisms will be performed to estimate the margin-to-failure of the lower head. A final project report that integrates the results of all the sample examinations and analyses will be issued at the completion of this program in June 1993. The results of this project, along with the results of previous TMI-2 studies, will contribute to an increased understanding of core melt sequences and the impact of such sequences on reactor vessel behavior.

The Smithsonian Institution is planning to exhibit tools and equipment that were used to extract samples from the TMI-2 reactor vessel under the VIP. The TMI-2 extraction tool exhibit will be included in a section on nuclear science that is planned as part of a major permanent exhibit on Science in American Life at the Smithsonian's National Museum of American History. Equipment for this exhibit may include the spare MDM cutting tool and articulating arm, practice cutting samples, a cut and polished metallographic specimen from one of the samples, protective suits used during the sample extraction (all uncontaminated), and a videotape taken during the actual extraction process. Current plans are for the exhibit to open around April 1994.

Instrumentation and Controls Technology Study

A study of the instrumentation and controls (I&C) technology used in nuclear power plants in Europe was conducted recently by a panel of U.S. specialists. This study included a review of the literature on the subject, followed by visits to some vendors, utilities, nuclear power plants, and research organizations in Europe that are leaders in the field of nuclear I&C.

Leo Beltracchi of the Office of Research was on the panel. Their study was sponsored by the National Science Foundation, and their findings have been published by Loyola College in Maryland as "European Nuclear Instrumentation and Controls," which is available from the National Technical Information Service.

Nuclear power plants in Europe, Canada, Japan, and the United States are moving toward the use of digital computers, especially microprocessors, for information and control systems. The amount of automation and the role of the operator are under discussion in all countries. In Japan and Germany, plants are moving toward a high degree of automation, whereas in France the emphasis is on computer-generated procedures with the decision to enable being made by skilled operators. Some Russian plants use digital systems to help the operator identify problems, decide on the appropriate corrective actions, and aid in the execution of these actions.

The panel made a qualitative comparison of the standing of U.S. nuclear power plants relative to the countries visited for status and progress in basic research, advanced development, and product implementation in seven categories: control room design, analog-to-digital transition, fault management systems, control strategies, I&C architecture, instrumentation, and standards and tools.

The panel concluded that European plants are ahead of the United States and moving further ahead in all seven categories, with the possible exception of instrumentation. European plants are also ahead in advanced development except for architecture and instrumentation. In basic research, European plants are ahead in four of the seven categories; however, the United States is about equal to Europe in instrumentation and in analog-to-digital transition, and the United States is ahead in architecture in general. In other words, U.S. computers are being purchased and used in all countries that the panel visited, but the development and implementation of the computers for nuclear power plant I&C is more advanced in Europe and Canada.

It appeared that the United States is behind in the development of digital systems for nuclear plants as well as in experience in using them. France has the most experience with digital safety systems and has built successful automatic control and informational systems, with the original design purchased from U.S. reactor vendors.

Germany has developed a unique control and safety strategy that automatically moves reactor systems back into the safe operating region. This automatic action minimizes the number of scrams, smoothes transients to minimize component stresses, and provides time for operator diagnoses for a broad spectrum of control failures, from both equipment failure and human errors. The system presently involves full digital reactor control, but for prevention and mitigation it uses semiconductor-based analog equipment, which is to be replaced by digital equipment without much increase in function.

Europe is ahead in the use of fault diagnosis and signal validation systems. Work on the use of digital information programs for fault management systems in nuclear plants is moving more rapidly in both Europe and Japan than in the United States.

The hardware for the digital systems used in all countries is by and large from U.S. computer companies, but limited deployment of digital systems in U.S. nuclear plants has curtailed the accrual of experience in the computer system architecture for I&C systems.

In France and Germany, control strategies have been extended to allow nuclear power plants to be used for automatic changes in power to match demands from the utility grid. These capabilities have improved control of local power distribution changes during transient power conditions.

In all countries visited by the panel, instrumentation for nuclear power plants is similar to that in the United States. Some special instrumentation is being developed; for example, a special neutron detection system is under development in France to provide improved in-core and ex-core power density and transient power level information. Germany has pioneered the use of prompt, in-core cobalt detectors for gathering detailed power density information.

The European countries are ahead in the use of computer-assisted software engineering tools, and they are more advanced in the development of standards. Standards and guidelines are the basis of the design and development of computer-based safety systems. The U.S. nuclear industry does not now have equivalent standards and guidelines for the development of computer-based safety systems; however, an effort to develop equivalent standards is under way.

An important point brought out in the survey is that the United States has been able to learn from the mistakes and overcomplexities of other countries. In France and Canada, the programmability of the digital systems enticed the users to add complexities that evolved into problems. Efforts must be made to maintain simplicity in systems, and problems are often not recognized until the review, quality assurance (verification and validation), and final approval stages.

Workshop on Nuclear Power Plant Aging and Life Extension in Moscow

During October, NRC staff and representatives of the Department of Energy and national laboratories participated in the US-CIS Joint Coordinating Committee on Civilian Nuclear Reactor Safety (JCCCNRS) meeting and the workshop on nuclear power plant aging and life extension.

Maintaining the safety of the older operating nuclear power plants in the Commonwealth of Independent States (CIS) was a major concern of the CIS participants. The Russian Federation (RF) places significant importance on the activities of this group, as evidenced by the active participation of the representatives of various institutions and organizations of the CIS as well as the operating plant personnel.

Information exchange through this group over the past 18 months has provided a foundation for developing both near-term and long-term programs for managing aging in their operating nuclear power plants with the primary emphasis on safety. It was recommended that the group concentrate on specific technical issues such as metal fatigue under operating environmental conditions, in situ degradation monitoring of naturally aged cables, and nondestructive examination.

How Clean Is Clean?

Chris Daily, DRA/RPHEB

NRC licensees who need to decontaminate land and structures as part of the decommissioning and license termination process must have criteria to determine "how clean is clean enough." The NRC must be assured that public health and safety and the environment are protected and ensure that the total dose to an individual is less than the public dose limit of 100 mrem/year. In addition, the NRC has set a goal for public doses attributable to residual contamination after decommissioning at a fraction of the public dose limit.

The dose that an individual might receive from residual radioactivity remaining at a site after decommissioning is estimated by first developing scenarios to describe potential future uses of the site. The scenarios allow possible exposures to be modeled and associated doses to be estimated. The modeling and scenarios can become extremely complicated, depending on the complexity of the site and the level of uncertainty that is

considered acceptable in the dose estimate. Detailed modeling may often be beyond the technical and financial capabilities of a large number of licensees – especially small licensees with limited resources.

NUREG/CR-5512, "Residual Radioactive Contamination From Decommissioning" (October 1992), has been developed to provide generic and site-specific dose conversion factors for residual radioactivity based on generic modeling. These dose conversion factors are used in a screening analysis to determine whether a site meets the criteria for release or whether more detailed analyses must be performed. The first volume of the report presents the scenarios, models, mathematical formulations, assumptions, and justification of parameter choices. It also includes responses to comments from the January 1990 draft of the report. The second volume will be a user's manual for an associated microcomputer-based program, and it will include tables of the generic dose conversion factors, example calculations, and the computer code listing. A third volume will contain a sensitivity analysis of parameters used in the modeling and a comparison with previously used guidance such as Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors."

The scenarios used in this generic analysis are prudently conservative but not necessarily bounding or "worst case." Selection of a prudently conservative scenario requires a great deal of professional judgment and common sense. The intent is to account for the vast majority of potential uses of lands and structures and overestimate the most probable annual dose while discounting a small fraction of highly unlikely uses that would result in higher doses. The prudently conservative approach does not include low probability scenarios that may result in higher calculated doses but are based on aberrant behavior or unpredictable and highly unlikely circumstances. The alternative was to use scenarios that would yield an upper limit on doses, i.e., bounding or "worst case," but would unnecessarily limit the usefulness of the resulting release criteria without providing significantly increased benefits to the public health, public safety, or to the environment. Hence, the dose conversion factors in NUREG/CR-5512 are judged to be higher than (i.e., overestimate) the most probable annual dose but may be lower than (i.e., underestimate) the bounding annual dose.

Licensees have some flexibility when applying the modeling contained in NUREG/CR-5512. If slightly increased accuracy or realism of the screening dose conversion factors is desired, and there is adequate justification, the generic (default) parameter values may be replaced with site-specific parameters. Within the modeling framework of NUREG/CR-5512, such a substitution of parameters would lead to site-specific derived dose conversion factors. The site-specific dose conversion factors may then replace the generic dose conversion factors in the screening analysis.

As another example of the flexibility of the approach, the NRC staff is developing a supplemental technical rationale and method for incorporating independent ground-water models into the NUREG/CR-5512 methodology so that a hierarchy of screening can be implemented. Existing groundwater flow and radionuclide transport models used in the performance assessment of low- and high-level waste facilities will be examined as possible tools in this methodology, and their appropriateness for screening sites with residual radioactive contamination that exceeds the "water use models" presently stated in NUREG/CR-5512 will be evaluated. The development of a hierarchical strategy to choose the appropriate groundwater flow and transport model, given the site radiological survey inventory and conditions, will also be performed.

NUREG/CR-5512 is one part of a large program the NRC staff has under way to provide information and guidance for the implementation of release criteria for the decommissioning of lands and structures. For example, NUREG/CR-5849, "Manual for Conducting Radiological Surveys in Support of License Termination" (Draft, June 1992), provides information on acceptable measurement and survey techniques and procedures. The draft manual is designed to provide the

licensee with guidance on planning, conducting, and documenting site surveys that can be used to demonstrate that a site has been decontaminated to a level consistent with the Commission's criteria. It contains specific guidance on the role of surveys in the decommissioning process, survey planning and design, selection and use of radiological instrumentation, conducting the survey, interpreting survey results, and documenting and reporting survey results. It also includes a sample Survey Plan and a sample Final Status Survey Report prepared in accordance with the procedures contained in the manual.

ASME Award to Bob Bosnak

Robert J. Bosnak of the Division of Engineering has been elected an Honorary Member of the ASME Boiler and Pressure Vessel Committee (B&PVC). Bob received this award in Anaheim, California, at the November 1992 meeting of the ASME Council on Codes and Standards. Bob served on the Committee for more than 20 years, first while working for the Coast Guard and then for the AEC/NRC. He recently resigned from the Committee as he plans to retire from the NRC in February 1993. Bob's efforts greatly contributed to the use of ASME codes and standards by both Federal agencies as a means of satisfying regulatory requirements. Bob's know-how and appreciation for the credibility of the voluntary standards programs have resulted in genuine respect from all who have served with him on the Committee.

Expert Statistical Review

The NRC deals with statistical issues in many forms all the time. In a recent letter to the Chairman of the NRC, the Advisory Committee on Reactor Safeguards emphasized the importance of expert statistical review of NRC products involving a statistical analysis. RES has an expert statistician on staff, Lee Abramson of the Division of Safety Issue Resolution, who also serves as an in-house statistical consultant to other offices within the NRC. His role is to detect and correct any statistical deficiencies or problems in NRC products, including contractor reports. His recent assignments include the following:

- In 1986, TVA's application for an operating license for Watts Bar was suspended when, among other problems, numerous defects in the QA records were found. In 1988, TVA instituted a Corrective Action Program (CAP) to correct the problems with the QA records. The goal of the CAP is to demonstrate that each of the 186 ANSI record types has no more than 5 percent defects (e.g., missing records) with 95 percent confidence. To achieve this goal, TVA proposed a sampling plan based on a Bayesian approach.

  There were two major flaws in the sampling plan. The first concerned the Bayesian approach used and the second dealt with the rectification procedure if defects were found. While a Bayesian approach based on subjective expert judgment is extensively used in nuclear probabilistic risk assessment (PRA), the situation at Watts Bar is different. First, contrary to the usual PRA situation in which expert judgment must be used because adequate data is unavailable, unlimited data is available at Watts Bar simply by sampling the QA records. Second, the goal in a PRA is usually to estimate a quantity (e.g., core damage frequency); at Watts Bar the goal is to test a hypothesis. This difference is crucial. The goal of a PRA is to gain insight into the risk; the goal of the Watts Bar CAP to is make a decision. Because the Watts Bar sampling plan can be very sensitive to the subjective choice of a prior distribution for the fraction defective (a necessary starting point for a Bayesian analysis), the NRC would not be able to validate the Bayesian approach. Instead, a classical statistical approach to meet the 95/5 acceptance criterion was recommended.

  A second flaw in the sampling plan concerned rectification. In order to minimize the required sample size, TVA opted to use a sampling plan with an acceptance number of zero. In this plan, a random sample of 60 is

chosen and the population is accepted only if no defects are found. In order to satisfy the 95/5 acceptance criterion, the population must be rejected if any defects are found in the sample. However, to avoid rejecting the population outright if a defect is found, TVA proposed to "rectify" the population by removing all records with the same type of defect from the population as well as from the sample. An additional random sample is drawn to replace the defects found in the sample of 60 and the population is accepted if no further defects are found. While this rectification procedure is intuitively plausible, it turns out that it will satisfy the 95/5 acceptance criterion only if the defect types to be rectified are completely specified before the initial sample is drawn. If this condition cannot be satisfied (which it rarely can be in practice), the sampling plan must be revised by increasing the sample size.

In response to the NRC position paper criticizing their CAP, TVA scrapped their Bayesian and rectification approach and adopted an approach based on classical statistics. NRC has accepted the revised CAP.

- Under current regulations, the containment of high-level radioactive wastes in the proposed Yucca Mountain repository is required to be "substantially complete" for a period of 300-1000 years, with the precise period to be chosen by the Commission at a later date. NMSS is considering how to interpret this qualitative rule in a quantitative fashion. They have tasked the Center for Nuclear Waste Regulatory Analyses (CNWRA) to delineate the technical considerations involved in substantially complete containment (SCC), to identify and develop uncertainty evaluation methods for SCC analysis, and to identify alternative regulatory positions.

  Three reports were produced for this task, including NUREG/CR-5639, "Uncertainty Evaluation Methods for Waste Package Performance Assessment." This document presents a quantitative framework for defining SCC in terms of an upper bound on the number of failed waste packages and a confidence level for attaining the bound. In order to deal with the large scientific uncertainties involved, NUREG/CR-5639 discusses a number of methodologies for uncertainty analysis. For this study, a new method was developed for combining bounds on one subset of input variables with distributions on another subset of input variables to produce bounds on the distribution of the output variables.

  In a report entitled "'Substantially Complete Containment' Feasibility Assessment and Alternatives Report," the CNWRA identified four alternative regulatory positions based on the quantitative framework developed in NUREG/CR-5639. One alternative is based on the current qualitative rule, one is based on a modified qualitative rule, and two are based on quantitative versions of the rule. All alternatives would require detailed technical guidance to DOE. In order to provide NMSS with insight as to which regulatory position to recommend, a prioritization exercise based on the methodology developed to elicit expert judgment for NUREG-1150 was carried out. A set of objectives and attributes to rate the alternatives was developed, and a panel of nine NRC staff members was elicited to rate the alternatives. To the surprise of some panel members, seven of the nine panelists preferred the quantitative alternatives. A description of the prioritization study and its results was published as CNWRA 92-016, "'Substantially Complete Containment' (SCC) Elicitation Report," in August 1992.

- The Radiation Protection and Health Effects Branch of RES is developing a "Manual for Conducting Radiological Surveys in Support of License Termination," NUREG/CR-5849, prepared by Oak Ridge Associated Universities. The manual uses a statistical hypothesis

testing procedure to determine whether the guideline value is exceeded and additional cleanup is required. For a set of measurements in a survey area, the sample mean plus an appropriate multiple of the standard deviation is compared with the guideline value. If any of the measurements is a localized "hot spot" (i.e., an area with elevated radioactivity), the procedure may be overly conservative and lead to requiring a cleanup for a survey area whose mean activity level is below the guideline value. A less biased estimate of the mean activity level can be obtained by using a weighted average (instead of the sample mean) in which the weights are proportional to the areas of elevated activity.

- As a part of the structural aging research program, the Structural Seismic Engineering Branch has produced a report on "Nondestructive Evaluation of the In-Place Compressive Strength of Concrete Based Upon Limited Destructive Testing," NISTIR 4874, prepared by the National Institute of Standards and Technology (NIST). The report takes available nondestructive concrete testing data as reported in the open literature and attempts to establish a correlation between nondestructive and destructive testing (which is more accurate) for residual concrete strength. Since the amount and quality of the available data were limited, a complex statistical procedure was used to establish the required correlation. The correlation was validated by splitting the data sets in half and checking each data set half against the other half. It is expected that the recently released NIST report will generate considerable interest in the civil engineering community.

- DOE has submitted a study plan for site characterization studies to calculate P_{rd}, the probability of magmatic disruption of the Yucca Mountain candidate repository site. Several statistical problems with the proposed study plan were identified, including:

  - The proposed methodology for calculating P_{rd} was based on an inappropriate use of regression plots and their associated confidence bands. Therefore, it was recommended that DOE consider developing a model for the spatial distribution of volcanic centers in the vicinity of Yucca Mountain and a model for the probability of repository disruption as a function of the location of a volcanic center, and then combine these two models to calculate P_{rd}.

  - In order to handle the uncertainty as to which of several competing models applies, DOE proposed using a weighted average, with the weights to be chosen by expert opinion. This approach is intended to reduce bias and to provide a summary assessment useful for regulatory decisionmaking. However, bias is not necessarily reduced by weighting alternative models. If the correct result lies somewhere in the middle of the model results, weighting models would tend to reduce bias. Nevertheless, there is no assurance that the correct result is in the middle – it might be at one of the extremes or even beyond. If this were the case, weighting could give a false assurance that bias had been reduced. In order that the results reflect the full range of scientific uncertainty as expressed by the experts, it was recommended that the methodology developed for NUREG-1150 be used as a basis for eliciting and aggregating expert opinion.

    In addition, using weighted models may destroy information essential for a proper regulatory decision. Suppose that Model A leads to a release that exceeds the regulatory requirements while Model B leads to a release that does not exceed the requirements. Suppose further that half of an expert panel prefers Model A while the other half prefers Model B, so that each is given equal weight. If the

weighted release does not exceed the regulatory requirements, the NRC may be led to believe that the repository design is acceptable. Since only one model, A or B, can be correct, this conclusion is justified only if there is high assurance that Model B is the correct model. However, since half of the experts prefer Model A, there is no such assurance. In this situation, the NRC's conclusion should be based on the realization that neither Model A nor Model B can be ruled out. Accordingly, it was recommended that the model results be presented separately, together with their associated weights and the accompanying justifications. This will allow the regulatory decision to reflect the full range of scientific uncertainty.

20th Water Reactor Safety Information Meeting

RES held its 20th Water Reactor Safety Information Meeting at the Bethesda Marriott Hotel on October 21-23, 1992. The meeting was attended by more than 500 participants representing the U.S. Government, the nuclear industry, and foreign governments. Commissioner E. Gail de Planque was the keynote speaker, and Eric Beckjord, Director of the Office of Nuclear Regulatory Research, spoke about NRC research, accomplishments, and prospects.

Eighteen sessions were scheduled in which researchers presented findings in such areas as severe accidents, primary system integrity, thermal hydraulics, advanced reactors, human factors, advanced control systems technology, aging, earth sciences, probabilistic risk analysis, and structural and seismic engineering. Also included was a special session presented by the Electric Power Research Institute (EPRI) on its nuclear safety research and development programs. The proceedings will be published early in 1993.

Modification of the ROSA Large-Scale Test Facility for AP600 Confirmatory Safety Testing

Gene S. Rhee, DSR/RPSB

Westinghouse Electric Corporation has submitted the Advanced Passive 600 MWe (AP600) nuclear power plant design to the NRC for design certification. The Office of Nuclear Regulatory Research is conducting confirmatory testing of AP600 safety systems. Such confirmatory testing is conducted to help the NRC staff evaluate the safety of the AP600 reactor systems.

In contrast to the current generation of reactors, this new design features passive safety systems for mitigating accidents and operational transients. Since these passive safety systems rely on gravity-driven flow, the driving forces for the safety functions are small compared to those available under conventional pumped systems. Thus, the performance of these new safety systems may be adversely affected by small variations in thermal hydraulic conditions. Also, the operation of the passive safety systems poses challenging computational problems for current thermal-hydraulic system analysis codes in that the current codes were not sufficiently assessed for conditions of low pressure and low driving heads and for the multiple flow paths used in the AP600 design. Therefore, a full-height, full-pressure integral effects test facility was needed for confirmation of AP600 safety system performance and for independent assessment and validation of computer analysis codes.

For confirmatory testing, it was determined that the most cost-effective route was to modify an existing full-height, full-pressure test facility rather than build a new one. Thus, all the existing integral effects test facilities, both in the United States and abroad, were screened to select the best candidate. The criteria for the initial screening included the size, facility configuration similarity, availability schedule, willingness to share the cost, and the ability to enter into a confidential agreement with Westinghouse for handling proprietary

information. This screening revealed that the best candidate was the Rig of Safety Assessment (ROSA) Large Scale Test Facility in Japan. To confirm these initial results and to determine the extent of modification necessary to simulate the AP600, the Idaho National Engineering Laboratory (INEL) was contracted to perform a comparative study between ROSA and AP600 using the RELAP5/MOD2.5 code. This study was published as NUREG/CR-5853, "Investigation of the Applicability and Limitations of the ROSA Large-Scale Test Facility for AP600 Safety Assessment" (December 1992); the main points of the study are presented here.

Existing ROSA Facility

The ROSA test facility is located at the Japan Atomic Energy Research Institute (JAERI) in Tokai-Mura, Japan. The existing facility, called ROSA-V LSTF, is a 1/48 volumetrically scaled, full-height, full-pressure conventional Westinghouse four-loop pressurized water reactor (PWR) simulator. The reference PWR used for the ROSA-V facility design was very similar to the Trojan Plant. When compared to AP600, ROSA-V represents 1/30-volume scaling. The ROSA facility includes two primary loops, each containing one cold leg, one hot leg, an active inverted-U tube steam generator, and an active reactor coolant pump. Each ROSA-V loop represents two of the reactor loops lumped together. The loop horizontal legs are sized to conserve the scaled volume as well as the ratio of length to the square root of diameter, L/D0.5, in order to correctly simulate the two-phase flow regime transitions. The inverted-U tube steam generators are full length and contain 141 tubes. Tube thickness, outside diameter, and length are identical to those of the reference PWR. A pressurizer is connected to one of the hot legs. The ROSA-V vessel includes an annular downcomer and contains 1064 full-length electrically heated rods capable of operating at 10 MW, or 14 percent of scaled full power for the reference PWR. The heater rod dimensions and pitch are the same as for the 17x17 fuel assembly used in the reference PWR core. Emergency core cooling (ECC) components, typical of those in the reference PWR, are included in ROSA-V. The current ROSA-V facility is very similar to the ROSA-IV facility described in a January 1989 JAERI report, JAERI-M-84-237, "ROSA-IV Large Scale Test Facility System Description."

Facility Modification

A comparison between the existing ROSA facility and the AP600 design showed that ROSA did not contain the key components important for safety response of the AP600. It was not obvious how much hardware modification to the ROSA facility would be needed to simulate the AP600. The fidelity of simulation must be balanced against the associated cost. The fidelity should be high enough to result in a facility capable of producing data for code assessment covering the major AP600 phenomena in the correct sequence. At the same time, the cost and the schedule have to be affordable. To make an optimum choice, INEL was asked to consider four levels of modifications in progressively more extensive stages. The first level of modifications was the absolute minimum, and the fourth level was the most inclusive among the four levels. To judge the fidelity of simulation of each level of modification, the following steps were followed.

Criteria Used for Evaluating Each Level of Modification

In evaluating each level of modification, the RELAP5/MOD2.5 code was used as a primary tool for comparing the predicted behavior of ROSA with that of AP600 for selected accident scenarios. This approach is based on the assumption that RELAP/MOD2.5, although not assessed against AP600 systems test data, will show major trends in overall behavior in such global parameters as depressurization rate, mass inventory, and energy distribution. The validity of this assumption is partially supported by the fact that the RELAP5 code reasonably matched experimental data from many different facilities, of different sizes, which were designed to simulate current PWRs. Since the thermal-hydraulic processes involved in

current reactors and passive reactors are fundamentally the same, it is likely that the RELAP5 code will also show the major trends in AP600 and ROSA, even though the predictions may not be as accurate until further improvements are made in such areas as mathematical modeling of condensation in the presence of noncondensible gases, boron transport, and the computation of level tracking and thermal stratification in a tank.

Accident Scenarios

In determining the ability of the ROSA facility to simulate the AP600 reactor, the following accident scenarios were analyzed with RELAP5 in both the AP600 and ROSA.

- A 3-in. diameter break in a cold leg
- A 1-in. diameter break in a cold leg
- A 3-in. diameter break in a pressure balance line between the core makeup tank (CMT) and a cold leg
- One and three tube ruptures in a steam generator
- A main steam-line break

These transients were selected because they challenge the passive safety features of the AP600. The processes and governing mechanisms participating in these transients span a reasonably complete range of important phenomena.

Different Levels of Modifications

The four levels of facility modifications that were considered are defined below.

I. First-level modifications were determined merely by inspection of the two designs, with only essential modifications considered, including the addition of the passive safety features not present in ROSA: the core makeup tank (CMT) and appropriate pressure balance lines, a passive residual heat removal system with simulated secondary cooling, automatic depressurization system with stages 1 through 3 on top of the pressurizer, stage 4 on the hot leg, and minimization of the pump loop seal lengths.

II. Second-level modifications were derived from the analysis of the first-level modifications and included the addition to the first level of a properly scaled AP600 pressurizer, surge line, and surge line connection.

III. Third-level modifications included all the above plus the splitting of one cold leg into two to incorporate two CMTs. A CMT is connected to each split part of the cold leg, as in the AP600.

IV. The fourth-level modifications resulted from the initial analyses, more in-depth inspection of the plant design differences, and discussions with representatives of the Japan Atomic Energy Research Institute, who owns the ROSA facility. These included the first and second levels coupled with appropriate upper head flow paths and adding an in-containment refueling water storage tank and two CMTs. Since there is only one cold leg in each loop, CMT cold leg pressure balance lines are connected to the same cold leg for most transients when asymmetry between the two CMTs is not expected, but connected to a different cold leg for a non-symmetric pressure-balance-line-break scenario.

The comparisons among RELAP5 calculations for different levels of modifications showed that the first-level modifications were capable of reasonably representing AP600 behavior during the early portion of most transients when asymmetric behavior between the two CMTs was not expected. The behavior in slow transients, or the latter part of fast transients, was distorted partly because of the larger friction and metal mass to volume ratio used in the calculations and partly because of the other differences in hardware, most of which were

eliminated as the Level I to Level IV modifications were made.

Since the first-level modifications have only one CMT, it can not simulate a situation in which two CMTs act differently, e.g., a break in the pressure balance line to one of the CMTs. On the other hand, splitting a ROSA cold leg into two to be able to attach a CMT to each part of the split cold leg (Level III modification) did not produce good results because, unlike AP600, the split cold legs had to be merged before they enter the vessel since another large hole could not be drilled into the vessel wall. Therefore, in the Level IV modification, splitting a cold leg was not incorporated. Instead, both of two CMTs were connected to the same cold leg when asymmetry between the two CMTs was not expected, and one of the two CMTs is connected to a different loop when asymmetry is expected. This arrangement produced reasonable approximation of the behavior of two CMTs in AP600.

Conclusion

Level IV modifications resulted in an acceptable fidelity of AP600 simulation in terms of overall system behavior as judged from the global parameters such as depressurization rate and liquid mass inventory. Further modifications were not considered to be cost effective even though there were additional differences that could have been eliminated, e.g., complete elimination of the primary coolant pump loop seal, different steam generator tube thickness, and different hot leg volumes. Level IV modifications are being implemented under a contract with Sumitomo Heavy Industries, which designed, constructed, and has been operating the ROSA facility as a contractor to JAERI. The facility modification is scheduled to be completed within a year, and a series of tests will be initiated in early 1994. The funds for testing will be provided by JAERI.

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