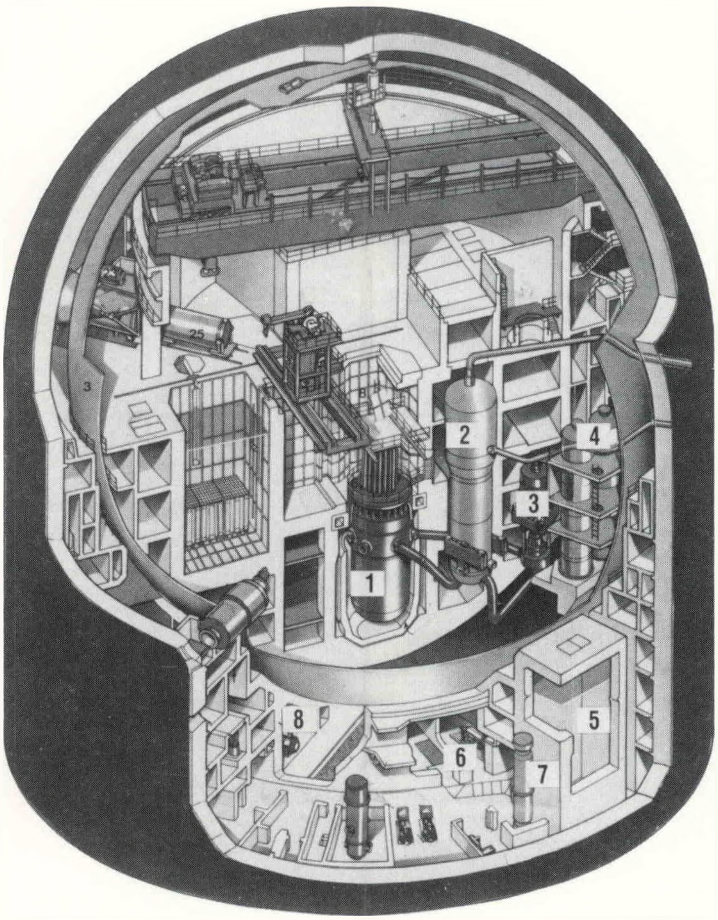
NUCLEAR SAFETY

VOL 34-2



TECHNICAL PROGRESS JOURNAL

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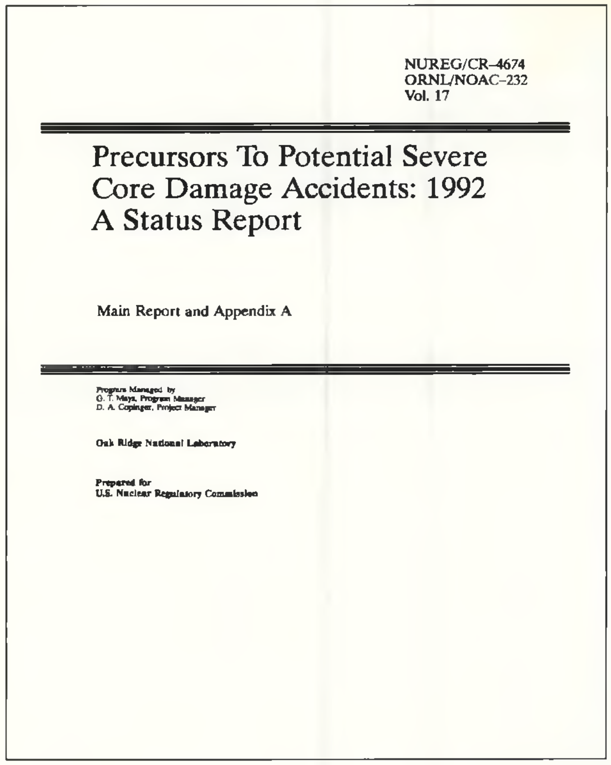

NUREG/CR-4674
ORNL/NOAC-232
Vol. 17

Precursors To Potential Severe Core Damage Accidents: 1992 A Status Report

Main Report and Appendix A

Program Managed by
G. T. Mays, Program Manager
D. A. Copinger, Project Manager

Oak Ridge National Laboratory

Prepared for
U.S. Nuclear Regulatory Commission

The Nuclear Operations Analysis Center (NOAC) of the Oak Ridge National Laboratory has prepared this latest member of a series of reports, whose coverage goes back to 1969, as part of its ongoing Accident Sequence Precursor Program. This program reviews licensee event reports (LERs) of operational events to identify and categorize precursors to potential severe core-damage accidents. Such precursors are infrequent initiating events or equipment failures that, had additional subsequent failures also occurred, could have resulted in a plant condition with inadequate core cooling. In other words, they are events that proceeded part-way on an identified path of multiple failures that could potentially lead to a severe core-damage accident but did not do so because the later failures did not occur. This report consists of Volumes 17 and 18 of the series; Vol. 17 contains the main report and Appendix A, and Vol. 18 contains Appendices B and C. This report is available from the National Technical Information Service, Springfield, VA 22161 or the Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20013-7082.

The Operational Performance Technology Section

The Operational Performance Technology (OPT) Section at Oak Ridge National Laboratory (ORNL) conducts analyses, assessments, and evaluations of facility operations for commercial nuclear power plants in support of the Nuclear Regulatory Commission (NRC) operations. OPT activities involve many aspects of facility performance and safety.

OPT was formed in 1991 by combining ORNL's Nuclear Operations Analysis Center with its Performance Assurance Project Office. This organization combined ORNL's operational performance technology activities for the NRC, DOE, and other sponsors aligning resources and expertise in such areas as

- event assessments
- performance indicators
- data systems development
- trends and patterns analyses
- technical standards
- safety notices

OPT has developed and designed a number of major data bases which it operates and maintains for NRC and DOE. The Sequence Coding and Search System (SCSS) data base collects diverse and complex information on events reported through NRC's Licensee Event Report (LER) System.

OPT has been integrally involved in the development and analysis of performance indicators (PIs) for both the NRC and DOE. OPT is responsible for compiling and analyzing PI data for DOE facilities for submission to the Secretary of Energy.

OPT pioneered the use of probabilistic risk assessment (PRA) techniques to quantify the significance of nuclear reactor events considered to be precursors to potential severe core damage accidents. These precursor events form a unique data base of significant events, instances of multiple losses of redundancy, and infrequent core damage initiators. Identification of these events is important in recognizing significant weaknesses in design and operations, for trends analysis concerning industry performance and the impact of regulatory actions, and for PRA-related information.

OPT has the lead responsibility in support of DOE for the implementation and conduct of DOE's Technical Standards Program to facilitate the consistent application and development of standards across the DOE complex.

OPT is responsible for the preparation and publication of this award-winning journal, *Nuclear Safety,* now in its 34th year of publication sponsored by NRC. Direct all inquiries to Operational Performance Technology Section, Oak Ridge National Laboratory, P.O. Box 2009, Oak Ridge, TN 37831-8065. Telephone (615) 574-0394 Fax: (615) 574-0382.

Cover: The cover shows a cutaway drawing of the 1300 MW(e) PWR of the German CONVOY series. This figure is taken from the article "R&D Activities on Safety Aspects of Future PWR Plants Performed at KfK" by B. Kuczera, which appears in this issue of *Nuclear Safety.* The numbered components are identified in Figure 1 of that article.

A quarterly Technical Progress Journal
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Nuclear Safety is a review journal that covers significant developments in the field of nuclear safety.

Its scope includes the analysis and control of hazards associated with nuclear energy, operations involving fissionable materials, and the products of nuclear fission and their effects on the environment.

Primary emphasis is on safety in reactor design, construction, and operation; however, the safety aspects of the entire fuel cycle, including fuel fabrication, spent-fuel processing, nuclear waste disposal, handling of radioisotopes, and environmental effects of these operations, are also treated.

Qualified authors are invited to submit articles; manuscripts undergo peer review for accuracy, pertinence, and completeness. Revisions or additions may be proposed on the basis of the results of the review process. Articles should aim at 20 double-spaced typed pages (including figures, tables, and references). Send inquiries or 3 copies of manuscripts (with the draftsman's original line drawings plus 2 copies and with black-and-white glossy prints of photographs plus 2 copies) to E. G. Silver, Oak Ridge National Laboratory, P. O. Box 2009, Oak Ridge, TN 37831-8065.

The material carried in *Nuclear Safety* is prepared at the Oak Ridge National Laboratory's Operational Performance Technology Section, which is responsible for the contents. *Nuclear Safety* is funded by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research. Editing, composition, makeup, and printing functions are performed by the DOE Office of Scientific and Technical Information (OSTI). Sale and distribution are by the U.S. Government Printing Office; see the back cover for information on subscriptions, postage, and remittance.

EDITORIAL

Major Changes in *Nuclear Safety*

It will scarcely have escaped notice that this issue of *Nuclear Safety* is being published with a very large delay, amounting to almost a year. I very much regret this delay and will try to explain why it happened and what will be the consequences for the future of this journal.

When *Nuclear Safety* began publication 34 years ago, it was funded by the Atomic Energy Commission, the forerunner of today's Department of Energy (DOE), and the Nuclear Regulatory Commission (NRC). Ever since these two agencies came into separate existences, the *Nuclear Safety* journal has been funded jointly by both. The relative amounts of support from the two funding organizations have varied over the years, but in the recent past DOE had supplied the major share of the funding. In mid-1993, as the result of, at least in part, severe restrictions in DOE's nuclear energy budget, DOE decided that it no longer wished to continue its financial support of the journal. A number of attempts from several directions to persuade DOE to reconsider this decision failed, and thus the journal was left with only the support from the NRC, which at the then-current level was not enough to continue operation in any meaningful way. Hence all work on the journal ceased in early fall of 1993, after already having had to suspend work for extended periods earlier in that calendar year. At that time issue 34(1) was out and issue 34(2) was only partly completed.

Fortunately, the NRC's Office of Nuclear Regulatory Research, under the leadership of its director E. Beckjord, then decided to request concurrence by the NRC Commissioners to increase the level of support of *Nuclear Safety* to the point where it can continue to be published, albeit with significant changes in scope and frequency of appearance. Because of the publication delay, this issue, Volume 34, No. 2, is presently planned to be the last one of Volume 34, to be followed by issue 35(1), though this matter still remains to be finally resolved. In fact, a number of questions concerning the frequency of publication and content of future issues still need to be decided, and I hope to inform you of their resolution in the next issue of *Nuclear Safety*.

In any event, I wish here to express my profound thanks and appreciation to Eric Beckjord and George Sege, his Administrative Assistant, who have shown enough faith in the usefulness and significance of this publication to go to bat to enable it to continue to exist and serve the nuclear community. We will certainly strive to be worthy of their efforts and support.

What will not change is the commitment of *Nuclear Safety* to continue to be a world-class journal dedicated to publishing both review articles and descriptions of important new work in the nuclear safety field from around the world. *Nuclear Safety* will continue to be an editorially independent, archival, peer-reviewed, professional technical–scientific publication covering the nuclear safety field.

For at least the next two years, however, it is likely that *Nuclear Safety* will appear only twice a year instead of quarterly. It will, nonetheless, publish considerably more than half the technical–scientific content per year in two issues than it had heretofore printed in four. This will be accomplished by eliminating or reducing the space devoted to the "current events" material, which has hitherto taken up about 30% of the journal.

As before, subscriptions to *Nuclear Safety* may be obtained from the Superintendent of Documents, U.S. Government Printing Office, Washington, DC 20402-9371.

Dr. Ernest G. Silver, *Editor-in-Chief*

Assessing Safety Culture

By L. Ostrom, C. Wilhelmsen, and B. Kaplan[a]

Abstract: *The concept of safety culture developed in the aftermath of the Chernobyl disaster. Researchers, however, have known for many years that safety performance is affected by an organization's socially transmitted beliefs and attitudes toward safety. The safety culture of an organization is very complex and hard to study, but it is possible to examine norms that make up the culture. A written survey instrument was developed to examine the safety culture of EG&G Idaho, Inc., a Department of Energy (DOE) Contractor at the Idaho National Engineering Laboratory (INEL). This instrument was developed by determining safety norms of the organization and then developing statements that reflect those norms for inclusion in the survey instrument. The survey instrument was used by DOE to assess the safety culture at INEL. Statistical tests on the data from the survey showed that the instrument had good internal consistency. The survey instrument, which is included in the article, appears to have merit for use by non-INEL organizations. This article also discusses how the survey should be administered and how the results can be used to help improve the safety culture of an organization.*

The purpose of this article is threefold. First, it discusses the concept of safety culture from a contemporary viewpoint. Second, it presents a survey instrument developed to assess the safety cultures of organizations. Third, it discusses how the results of the survey instrument can be used to improve safety culture.

[a]Idaho National Engineering Laboratory, EG&G Idaho, Idaho Falls, ID 83402. The views and conclusions in this article are those of the authors and do not necessarily reflect the policies of the U.S. Department of Energy.

DEFINITION OF SAFETY CULTURE

The concept of safety culture developed in the aftermath of the Chernobyl disaster.[1] However, the concept that the organization's beliefs and attitudes, manifested in actions, policies, and procedures, affect its safety performance is not new. In fact, Heinrich's *Domino Theory* developed in the 1930s was based on the premise that a social environment conducive to accidents was the first of five dominos to fall in an accident sequence.[2] The other four dominos in sequence were fault of person (personal traits), unsafe act, accident, and injury. This theory is now 60 years old, and much research has been done in this area since; however, from our discussions with managers and safety professionals, there is still a lack of understanding as to what safety culture is or how to assess it.

What is safety culture? The American Heritage Dictionary defines culture as "The totality of socially transmitted behavior patterns, arts, beliefs, institutions, and all other products of human work and thought characteristic of a community or population."[3] A culture is comprised of norms or patterns of perceptions, speech, and even building design features that make the culture what it is. It is difficult to understand a culture in total, but it is possible to study and understand individual norms. A social norm is defined as an unspoken rule of behavior that, if not followed, will result in sanctions. In an organization, a norm might be that managers wear suits. In this organization, a manager who arrives at a meeting in casual clothes might be teased or reprimanded. If he consistently failed to wear a suit, he might be considered unprofessional, not reflecting the company image, and face severe sanctions, including loss of his position.

What constitutes a safety norm, then? An example might be that in a company employees receive special recognition for reporting accidents. This could be considered a positive norm. Another example of a norm might be when individuals no longer seek solutions to safety concerns and stop looking to their safety professionals for help because they expect them to be unavailable. This might be considered a negative norm.

Pidgeon[1] says that a "good" safety culture is hard to define. Part of the reason for this is that each organization's culture is somewhat unique. Culture can be influenced by the nation or region, by the technologies and tools it uses, and by the particular history of success and failure it has achieved. Safety culture of an organization may be influenced by the marketplace and regulatory setting in which it operates. Safety culture may be influenced by the vision, values, and beliefs of its leaders as well. All these influences make it difficult to say what a "good" safety culture will look like in a particular setting.

Despite differences, good safety cultures do have things in common.[1] Good safety cultures have employees with particular patterns of attitudes toward safety practice. Because it is impractical to establish formal, explicit rules for all foreseeable hazards, norms within the organization are required to provide guidance in particular circumstances. In a "good" safety culture employees might be alert for unexpected changes and ask for help when they encounter an unfamiliar hazard. They would seek and use available information that would improve safety performance. In a "good" safety culture, the organization rewards individuals who call attention to safety problems and who are innovative in finding ways to locate and assess workplace hazards. All groups in the organization participate in defining and addressing safety concerns, and one group does not impose safety on another in a punitive manner. The result is an overall positive attitude toward safety.

Organizations with a "good" safety culture are also reflexive on safety practices. They have mechanisms in place to gather safety-related information, measure safety performance, and bring people together to learn how to work more safely. They use these mechanisms not only to support solving immediate safety problems but also to learn how to better identify and address those problems on a day-to-day basis.

What is acceptable in a company regarding safety must be defined and practiced if a corporate culture that values safety is to be created.[4] Ideally, employees should know all the risks associated with their jobs, what is required for safety, and take responsibility for themselves. In other words, develop a norm in which employees are aware of all the risks in their workplace or are continually on the lookout for risks.

ASSESSING SAFETY CULTURE

How does an organization assess its safety culture? A plan called the Safety Outreach System developed by John Thirion, corporate safety director at Johnson & Johnson, emphasizes asking employees what their safety concerns are and then responding to those problems.[4] "You start asking every employee, every visitor, every contractor, 'What worries you the most about your safety? What hazards do you see here in the work place? Where is the next accident going to occur? To whom? What can we do to prevent it?' What I do is create the most real time safety agenda that any management can have," says Thirion. This is a very desirable system. Also needed within the organization, however, is a means of measuring and comparing improvements or decrements in safety culture. We have found that a standardized written survey instrument can and should be used in addition to informal employee interviews to gain a broader understanding of the safety culture.

Bailey and Petersen[5] concluded that a safety perception survey is useful because (1) the effectiveness of safety efforts cannot be measured by traditional procedural-engineered criteria like safety reviews, audits, and inspections; (2) the effectiveness of safety efforts can be measured with surveys of employee perceptions; (3) a perception survey can effectively identify the strengths and weaknesses of elements of a safety system; (4) a perception survey can effectively identify major discrepancies in perception of program elements between hourly rated employees and levels of management; and (5) a perception survey can effectively identify improvements in and deterioration of safety system elements if administered periodically. We agree with the conclusions of Bailey and Petersen. In addition, a properly developed survey instrument can be a valuable tool to compare against a company's accident–illness record or to provide data in the form of survey results in safety meetings covering the real safety concerns that employees have. A survey can enable an organization to compare the results from a certain department or company with another in a standardized, structured manner that helps target efforts in light of limited safety budgets.

Currently, there are very few safety surveys cited in the literature. Bailey and Petersen[5] discuss the use of a perception survey to assess safety system effectiveness among four railroads. The survey instrument they used, however, was not presented in the article.

DEVELOPMENT AND VALIDATION OF THE SURVEY

Bruce Kaplan developed an original version of the safety norm survey in 1989. The development process included three techniques. The first technique involved interviewing 86 EG&G Idaho employees, including managers, professionals, office workers, and laborers from various facilities at the Idaho National Engineering Laboratory (INEL). The individuals were asked three interview questions addressing safety and procedure compliance at EG&G Idaho. These questions were:

1. Suppose that three years from now our company had become a national leader in safety. What would you see people doing with regard to safety?
2. For each of the major areas named, how far do you think we have to go from the way things are now?
3. For each area rated, what do you see going on now, or not going on now, that makes you say we have that far to go?

The first of these questions was designed to elicit desired future norms, the second question was intended to have people consider and compare the present with the desired future, and the third was designed to elicit current norms. Results of the interviews were content analyzed and used to generate several of the items in the survey.

The second technique used to generate survey items involved holding an all-managers meeting in which managers were asked to write down a personal safety credo: what they say they believe about safety that they would like each of their employees to understand. Examples of the managers' credos included the following:

I believe . . .
. . . .That safety is everyone's personal responsibility. It begins with a strong and aggressive management involvement and commitment. I believe it takes daily suggestions and interactions with the workforce to remind, improve, enhance, and reinforce the company's commitment to protect employees.

. . . .Safety is the result of behavior, modeled by top management and characterized by honesty; truthfulness; and patient, persistent, and purposeful concern over the well-being of every individual in our community. Safety must be developed into a social style.

The credos were content analyzed and sorted into themes or categories according to their subject matter. The categories developed were Individual Responsibility, Safe Processes, Safety Thinking, Safety Management, Priority of Safety, and Safety Values.

A third technique was used to ensure comprehensiveness of the survey instrument. This technique involved querying other sources of information, such as previous interview data concerning a recent organizational climate survey, a literature review, and previous personnel opinion surveys, for possible norms. Possible safety norms suggested by these sources were selected for inclusion in the new survey instrument. Review of the literature concerning organizational climate, organizational norms, safety climate, and safety norms provides a conceptual framework into which items might be organized. Of particular importance in this sorting was the research of Litwin and Stringer.[6] The categories of safety norms ultimately selected were very similar to their categories of social norms except that ours were particularly adapted to safety. The data gathered were sorted into the following categories: Safety Awareness, Teamwork, Pride and Commitment, Excellence, Honesty, Communications, Leadership and Supervision, Innovation, Training, Customer Relations, Procedure Compliance, Safety Effectiveness, and Facilities.

A total of 84 statements, divided among the categories, were included in the original survey. Statements on the survey instrument presented had both positive and negative wording. In general, positive wording was selected when interview data suggested a positive norm, such as "people work safely, even when the boss isn't looking." Negative wording was selected when interview data suggested a negative norm, such as "We hesitate to report minor injuries and incidents." An attempt was also made to have a reasonable balance between both positive and negative wordings. The completed survey instrument was then administered to 121 employees in 1989.

In December of 1990 the Department of Energy (DOE) decided to conduct a safety culture survey of the Idaho National Engineering Laboratory (INEL). The DOE selected the EG&G Idaho, Inc., survey instrument for this purpose. The survey was modified to include four additional statements. These statements were included to determine specific pieces of information desired by the INEL contractors. The survey was administered during the month of January 1991 to about 4000 employees of DOE-ID and its eight contractors (EG&G Idaho; Rockwell; MSE, Inc.; Chem-Nuclear Geotech; West Valley–Nuclear; Winco; PTI; and MK-Ferguson). A statistical sampling method was used that specified the number of employees needed to be surveyed to have a 95% level of confidence in the data. The results from the survey pointed out both the strengths and weaknesses in the safety cultures of the organizations. The survey was recently modified by Cheryl Wilhelmsen and Jerry Harbour, Ph.D., for use in helping to assess the safety culture at the Rocky Flats DOE site.

The Cronbach's Coefficient Alpha test was performed on the data from the 1991 administration of the survey to determine the reliability of the survey. The statistic, Cronbach's Coefficient Alpha,[7] has a range of zero to one. A low value indicates that the survey instrument–statement has little internal consistency and needs to be restructured. A high value indicates good internal consistency. A one indicates that the instrument–statement has perfect internal consistency and is currently perfectly structured. The analyses showed that the survey instrument had very good internal consistency with Alphas approaching 0.96.

Although it is difficult to determine whether the perfect balance of positively and negatively worded statements was made during the survey development process, the Pearson product–moment correlation coefficients[8] for individual questions with the total survey, for all questions except number 21, ranged from $r = 0.40$ to 0.67, which indicated reasonable correlations. The correlation coefficient for question 21 was $r = 0.15$, which indicated poor correlation. The range of Pearson correlation coefficients for individual questions within a group of questions (i.e., the safety awareness grouping) ranged from $r = 0.63$ to 0.83, which indicated good correlations. These results indicated that overall questions fit well into the survey as a whole and within the individual groups of questions.

We feel the EG&G Idaho safety norm survey has merit for use by industry outside the DOE system. Therefore the instrument itself is included as an Appendix to this article. The following discussion describes how the survey should be administered and how the results can be used to improve safety culture.

ADMINISTERING THE SURVEY

The context of survey administration is crucial. Research has shown that constructive changes only come about when feedback, analysis, and action planning are integral parts of the data collection and reporting process. The first step in the administration process is to decide who should be surveyed. Three questions can be asked to help make this decision. They are: (1) What level of statistical confidence is desired in the data? (2) Will employees feel neglected or become angered if they are not included in the sample population and the survey is not administered company-wide? (3) Are the people expected to take action on the results included in the survey sample?

A statistician should be consulted to help answer the first question, and an informal survey of employees can be conducted to help answer the second. Those individuals who will have to take action on the results of the survey should always be included in the survey sample in the same ratio as the rest of the working population. If any problems with administering the survey to a sample of the work population are detected, then the survey should be administered company-wide.

The employees who will be given the survey should be informed approximately a week before the actual survey administration. At this time they should be told the purpose of the survey and the survey process. The facility for completing the survey should be near the employees' actual place of work with adequate space for writing, bathroom facilities, and quality lighting. Also, there should be special provisions for employees who are physically handicapped and/or reading impaired. The survey should be given in groups of employees large enough so that employees feel anonymous but not so large that an employee who needs help is overlooked.

In conducting any type of research it is desirable to find out how each group of subjects responded to the lowest subdivision of the organization as possible. In this type of survey, however, individuals might bias their responses more positively if they felt a manager could determine what their personal responses were. If, for instance, the survey asked for job title, supervisory level, years in service, department, and educational level, it would be possible to pick out who that individual was. Employees know this and might answer their survey differently. To get good data, it is better to ask the fewest possible demographic questions and to restrict those to broad categories, such as department and supervisory level. The employees will feel more comfortable taking the survey. The company will benefit by getting better, more honest data.

The directions on the survey should again state clearly the purpose for the survey and how to complete it. The directions should also ask respondents to answer each statement for the company–organization as a whole or the part of the company–organization with which they are most familiar. They are specifically asked not to evaluate their own manager or work group. The purpose of this broader focus is to ensure the objectivity and reduce the defensiveness. It is also assumed that employee perception of norms in these broader settings would have significant impact on local settings.

Each statement in the survey instrument should be followed by a scale. The five-point scale allows respondents to indicate the extent to which they agree or disagree with each statement. An example of a scale is shown in Table 1.

Table 1 Example of Scale

Strongly disagree	Disagree	Neither disagree nor agree	Agree	Strongly agree
1	2	3	4	5

Responses 1, 2, 4, and 5 in Table 1 are self-explanatory; however, the third, neither disagree nor agree response, is not as obvious. If an employee responds with a 3, they are saying they are neutral in their response to the statement. This does not mean the item does not pertain to them; they are saying they do not have an opinion either positive or negative concerning an item. This is a legitimate response for an employee to have. The instructions should say that if a statement does not pertain to you then do not answer it. The data generated from individuals not responding to statements are also of significant value. The percent nonrespondents for a statement can give an indication of the employees' assessment of those questions which pertain to them. Sutton[9] says that nonparticipant data are important because they can give an indication that individuals (1) have never been asked to participate in the process being investigated or (2) cannot or are not willing to participate in the survey process. Therefore the reasons why individuals did not respond to statements should be investigated further.

USES OF THE SURVEY DATA

All available forms of data should be collected and analyzed before making judgments about the safety culture of an organization. In addition to the questionnaire itself, data gathered should include accident statistics, safety performance data, records of employee and management concerns, and other measures of product quality and organizational performance.

Other important sources of input to the analysis process are the explanations and interpretations given by those surveyed. Ideally, each group surveyed should be given an opportunity to review and interpret their own results and to provide input to those trying to draw inferences across many groups and organizations.

The following describes how the data can be used alone or in conjunction with other information to get as complete an understanding of the safety culture of an organization as possible.

Descriptive statistics is a collection of methods for classifying and summarizing numerical data.[8] Descriptive statistics include mean, median, percent nonrespondents, and frequencies of response. These can be displayed both in numeric form and using graphics, such as bar graphs. For the results of a survey such as this, graphical presentation of the data is the most logical. Someone looking at the results can rapidly scan the data and determine what topical categories and departments–organizations need attention. The following discussion pertains to the graphical portrayal of the data. Please note in these examples that the results of the negative statements have been reversed, so the desired response is now 5. Please note that these examples are based on real data but do not reflect the results of any one company.

Figure 1 shows the type of bar charts that can be developed. This chart shows the means for the statements within the Safety Awareness Section. The following are the statements that make up this section:

1. In our company, the employees are aware of their part in safety.

2. In our company, people think safety concerns do not relate to office workers.

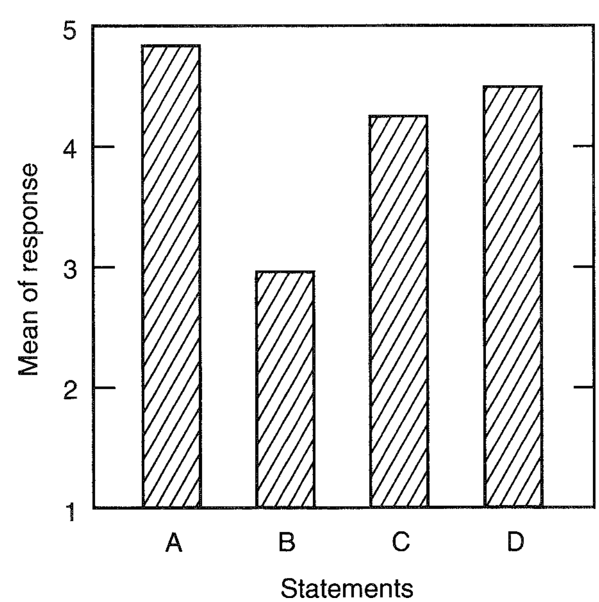

Fig. 1 Responses for the safety awareness section.

3. People are well aware of the safety hazards in their area and are careful to minimize and avoid them.

4. Around here, people don't think much about safety.

It is evident, looking at the responses to statements 1, 3, and 4, that employees are aware of their part in safety. Therefore an intervention designed to increase safety awareness may not be indicated. Statement 2, however, indicates that employees generally feel that safety concerns do not relate to office workers. If, in this setting, many office workers were injured each year, then this area would need attention.

Figure 2 shows how a group of departments responded to Statement 9, "Safety personnel are unavailable when we need help." Results from Departments B and E appear less positive than those from the other three departments. This may be a flag indicating that the perceptions about the safety personnel in Departments B and E are negative. Figure 3 shows the corresponding normalized accident statistics for those departments. Comparing these two figures, it appears that Department E may have a problem with its safety personnel, and this problem could be having an impact on employee safety. When we look at the results for Statement 5, "Safety professionals in this company tend to be bright and capable people" (Fig. 4), we again see that the results from Department E appear different from those from the other departments.

Can we then conclude that Department E has a problem with its safety personnel? To answer this question, the involvement of the people in Department E and the safety personnel that support them is required. Other people in the company who have been in a position to have observed Department E over time could also make a significant contribution to answering this question. Getting all these people involved, especially those who would be needed to design and implement a successful solution, might be a logical next step. By getting them all into one room to talk together about the issues might be the

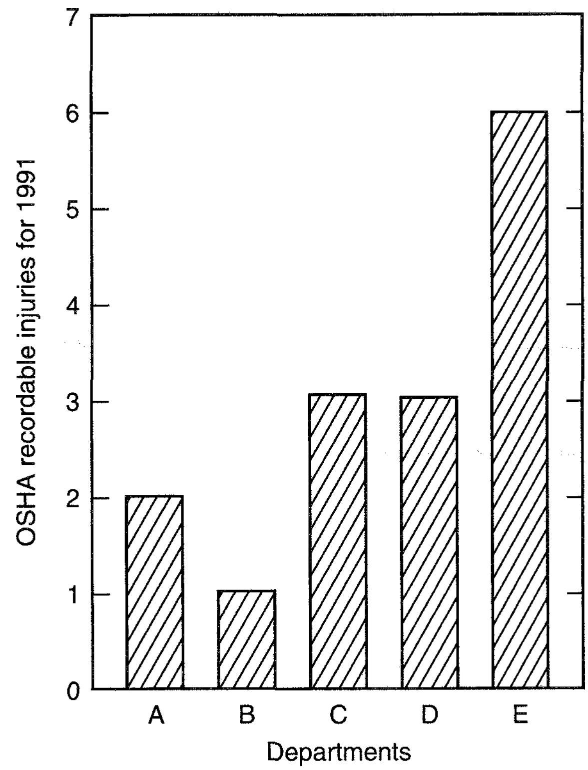

Fig. 3 Accident statistics by department.

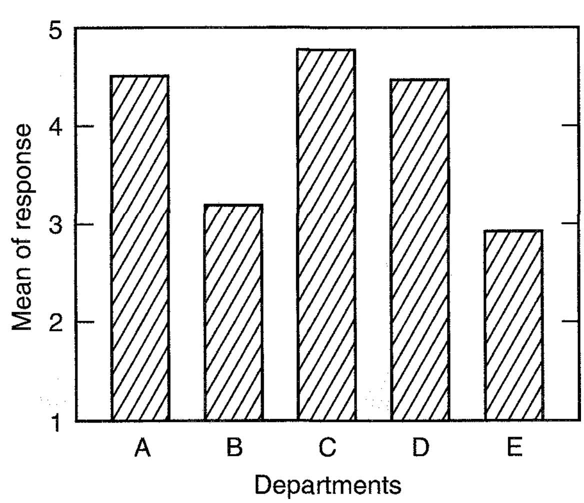

Fig. 2 Responses for statement 9 by department.

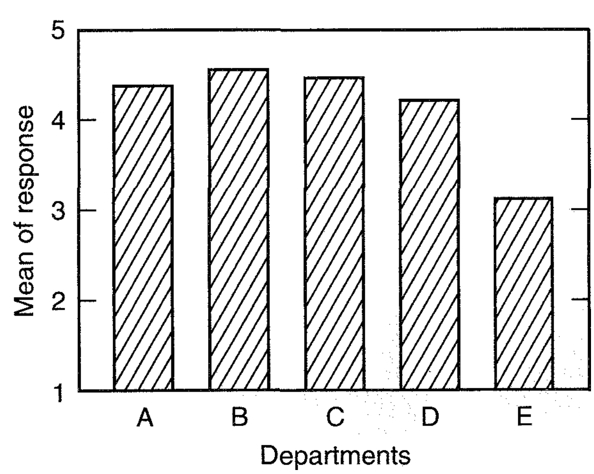

Fig. 4 Responses to statement 5 by department.

strongest move the safety professional can make to help understand and improve this situation.

Further diagnosis with the parties involved may show, for example, that it is not a problem with the safety personnel but with the number of personnel or a lack of pertinent experience or with the degree of hazard associated with the tasks in Department E. When making these comparisons it is important to compare only departments that perform similar types of work. A department that does only office work should not be compared with a warehouse operation.

Figure 5 shows a bar chart with the data broken down by percent negative (respondents who answered negatively), positive (respondents who answered positively), and neutral response and percent nonrespondents for statements 1, 32, and 42. It is evident that the overwhelming number of respondents answered Statement 1, "In our company, the employees are aware of their part in safety," in a positive manner. For Statement 32, "Timely feedback is seldom provided when a safety hazard is reported," however, there is a higher percentage of negative responses. This indicates that employees feel safety problems should be attended to in a more expeditious manner. The results from Statement 42, "In our company, employees who will implement plans are seldom involved in reviewing their safety implications," indicates that people may not know whether safety implications are always considered thoroughly. Also, the high percentage of nonrespondents may indicate that employees are not "on-board" in regard to considering safety.

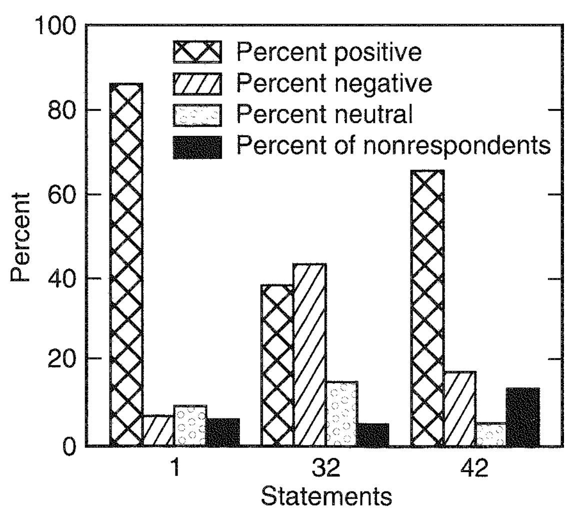

Fig. 5 Responses broken down by percent positive, percent negative, percent neutral, and percent of nonrespondents.

Inferential statistics, such as Student's t-test, chi-square goodness of fit test, and correlation analysis, can also be used to analyze the results of the survey. Although these are powerful tests and help to further elucidate the results of the survey, they are also much more difficult to interpret and, in this context, provide management with little more useful data than do the descriptive statistics alone.

As with all other aspects of a business, employees need to be involved with helping to interpret the data. Survey responses, at best, provide only an indication of what employees views might be. Properly presented, the responses can stimulate a focused discussion and exploration among employees and between employees, their management, and interfacing organizations. Survey responses can help the parties involved to identify for themselves some of their most important safety questions and can be used to stimulate productive inquiry into how to bring about improvements. A first step in this direction is to ensure that the results of the survey are communicated to the employees as soon as possible.

SUMMARY

By assessing its safety culture, an organization can determine where efforts need to be focused. Optimally, every employee should be involved in determining and addressing safety concerns. This, however, is not always possible. A properly structured survey instrument has been shown to be a very effective tool for assessing safety culture in organizations.[5]

Safety professionals should play a lead role in administration and analysis of the survey data. To achieve results, however, an organization needs to find ways to get the people who were surveyed to engage in reflection on what the data mean and what actions they can take to address the problems identified.

The EG&G Idaho Safety Norm Survey has been found to be an effective survey instrument with good internal consistency and has been used to assess the safety culture at several DOE facilities.

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APPENDIX

EG&G Idaho Safety Norm Survey

Safety Awareness

1. In our company, the employees are aware of their part in safety.
2. In our company, people think safety concerns do not relate to office workers.
3. People are well aware of the safety hazards in their area and are careful to minimize and avoid them.
4. Around here, people don't think much about safety.

Teamwork

5. Safety professionals in this company tend to be bright and capable people.
6. In this company, people ask for help with safety when they need it.
7. Around here, you'll be better off if you hide your problems and avoid your supervisor.
8. People do go out of their way to help each other work safely.
9. Safety personnel are unavailable when we need help.
10. Around here, employees who have to follow safety and health procedures are seldom asked for input when the procedures are developed or changed.

Pride and Commitment

11. Around here, people take pride in how safely we operate.
12. In this company, people stand up for the safety of their operations when others criticize it unfairly.
13. Around here, people look at the company safety record as their own safety record and take pride in it.
14. In this company, I cannot significantly impact the company's safety record.
15. In this company, people think safety isn't their concern—it's all up to their manager and others.
16. Around here, people see safety as the responsibility of each individual.
17. This company cares about the safety of its employees.

Excellence

18. In this company, we have the highest standards for safety performance.
19. Around here, people are always trying to improve on safety performance, even when they are doing well.
20. People are often satisfied with routine and mediocre consideration for safety.
21. Around here, the way we work now is safe enough.
22. In this company, there is no point in trying harder to be safe; no one else is.

Honesty

23. In this company, people work safely, even when the boss isn't looking.
24. Around here, people wear safety equipment even when they know they aren't being watched.
25. Around here, people are willing to comply with safety measures and regulations.
26. In this company, people try to get around safety requirements whenever they get a chance.

Communications

27. In this company, we hesitate to report minor injuries and incidents.
28. We don't get adequate information about what is going on with safety in the company.
29. Around here, there's lots of confusion about who to contact for safety concerns.
30. Around here, safety statistics are seldom studied and discussed.
31. In our company, safety hazards are seldom discussed openly.
32. Timely feedback is seldom provided when a safety hazard is reported.
33. In this company, you cannot raise a safety concern without fear of retribution.
34. In this company, we have very few safety signs or posters.
35. Around here, employee ideas and opinions on safety are solicited and used.
36. People who raise safety concerns are seen as trouble makers.

Leadership and Supervision

37. It's a tradition; safety matters are given a low priority in meetings.
38. In our company, managers don't show much concern for safety until there is an accident.
39. In this company, the people who make safety decisions don't know what is going on at the workers' level.
40. Around here, work is organized so that you can do the job safely.
41. Around here, managers seldom work with their groups to identify and correct safety concerns or problems.
42. In our company, employees who will implement plans are seldom involved in reviewing their safety implications.
43. Managers/supervisors are often not available to answer health and safety questions.
44. My manager/supervisor discussed safety and health issues in my last employee evaluation.
45. Supervisors are receptive to learning about safety concerns.
46. In this company, people who work safely get no real rewards.
47. Little special recognition is given to safe employees.

Innovation

48. Around here, people are constantly on the lookout for ways of doing things more safely.
49. People tend to hang on to the old ways of doing things without regard to their safety implications.
50. In this company, people are encouraged to express new safety ideas and suggestions.
51. Around here, you get little recognition for new safety ideas.
52. It's a tradition; you don't raise safety ideas that your boss doesn't have first.

Training

53. People mostly give lip service to safety training; they do little to actively support it.
54. In this company, safety training is compromised in favor of more pressing demands.
55. Around here, managers are not very well trained to identify and address safety concerns.
56. In this company, safety training doesn't address subjects of real concern.
57. It's a tradition; safety training is done on a regular basis.
58. People in this company are well prepared for emergencies, and everyone knows just how to respond.
59. I know who to talk to when I see a hazard or have health and safety concerns.

Customer Relations

60. Employees here are always looking for ways to satisfy the customers' needs and requirements.
61. Customers here count on our company to do its work safely.

Procedure Compliance

62. In this company, we have a long way to go in improving our compliance.
63. In this company, people are often uncertain about what the safety procedures are for the work they do.
64. In general, people are well acquainted with the safety procedures for their job.
65. In this company, the safety procedures are relevant to employees' particular circumstances.
66. Around here, there are lots of safety procedures that don't really apply to the particular areas or circumstances in which they are supposed to be used.
67. There are so many procedures they interfere with doing a job safely.
68. In this company, area requirements for protective clothing and equipment may not reflect the actual hazards.
69. In this company, employees use their heads and raise lots of questions about why things are being done the way they are.
70. In this company, procedures are too detailed, making compliance a mindless activity.
71. It's a tradition; people carefully follow the written procedures.
72. In this company, people can be confident they are safe when they are following the rules.
73. Around here, you can't expect praise and recognition for complying with procedures.
74. In this company, following safety procedures is consistently expected.
75. Safety procedures tend to be too vague and general to apply in specific situations.

Safety Effectiveness

76. When it comes down to it, people in this company would rather take a chance with safety than miss a schedule or budget commitment.
77. In this company, people are willing to expend a great deal of effort to get a job done safely.

78. In this company, work is not done that jeopardizes other workers or the public.
79. Employees rarely take the initiative to get safety problems taken care of.
80. Around here, people can report a safety problem several times, yet the problems may remain and not get corrected.
81. Our daily routines don't show that safety is an important value.

Facilities

82. In this company, the physical conditions of work locations inhibit safe work.
83. In this company, facilities are designed with safety in mind.
84. Concern and attention is being given to maintaining good safety conditions in our facilities.
85. People tend to keep their facility neat and orderly.
86. Around here, good housekeeping isn't just the janitor's job—people clean up their own areas.
87. In this company, fire and electrical hazards are accepted in some of our facilities.
88. Around here, we really keep on top of the snow and ice problems and prevent them from getting out of hand.

The U.S. Nuclear Regulatory Commission Thermal–Hydraulic Research Program: Maintaining Expertise in a Changing Environment

By B. W. Sheron,[a] L. M. Shotkin,[b] and A. J. Baratta[c]

Abstract: *Throughout the 1970s and early 1980s, the U.S. Nuclear Regulatory Commission's (NRC's) thermal–hydraulic research program enjoyed ample funding, sponsored extensive experimental and analytical development programs, and attracted worldwide expertise. With the completion of the major experimental programs and with the promulgation of the revised emergency core-cooling system rule, both the funding and prominence of thermal–hydraulic research at the NRC have declined in recent years. This has led justifiably to the concern by some that the program may no longer have the minimal elements needed to maintain both expertise and world-class status. The purpose of this article is to describe the NRC's current thermal–hydraulic research program and to show how this program ensures maintenance of a viable, robust research effort and retention of needed expertise and international leadership.*

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The safety performance of nuclear reactors in response to postulated accidents is determined almost exclusively by analyses using thermal–hydraulic system computer codes. Thus code development and improvement form one cornerstone of the U.S. Nuclear Regulatory Commission (NRC) thermal–hydraulic research program. Before these codes can be used with confidence, they must be assessed against data from scaled test facilities. This testing is done to ensure their accuracy for full-scale plant analysis; therefore performing tests to provide data for code assessment forms the second cornerstone of the NRC program. The assessed codes are then used by the NRC to provide an independent technical basis for regulatory decisions on nuclear plant design and operation. In a companion paper ("The USNRC Thermal–Hydraulic Research Program," L. Shotkin and D. Bessette, *Nuclear Engineering and*

Design, to be published) information is provided on the scope of the NRC's thermal–hydraulic activities. In this article we concentrate on the question of maintaining expertise in this important discipline and put forth the key elements for maintaining a robust, viable thermal–hydraulic analysis capability for the NRC:

1. *Challenging technical environment.* Technical excellence cannot evolve or develop unless there is a technical challenge to provide the motivation.
2. *Maintenance of a cadre of experts.* Technical excellence is strongly enhanced by the synergism associated with technical teams. Maintaining a cadre of thermal–hydraulic experts ensures this enhanced technical excellence. It is also recognized that support for universities ensures training of new potential experts.
3. *Assurance of adequate resources.* Technical excellence does not come for free. Adequate funding is necessary to establish the programs that provide the technical challenges.
4. *Continuing involvement with experimental programs and the international community.* The challenge for thermal–hydraulic experts is the ability to understand and predict reality. Involvement in necessary experimental programs, both domestic and foreign, is therefore essential to achieving technical excellence. Similarly, continuous interactions with the international community are essential elements of maintaining world-class thermal–hydraulic expertise.

In the following discussions, we first present a short history of the thermal–hydraulic program evolution at the NRC during the past 20 years. By knowing where the program has been and by recognizing the realities of the current regulatory environment, we can develop a practical and workable program based on the four key elements previously described.

THE CHANGING ENVIRONMENT FOR THERMAL–HYDRAULIC RESEARCH

In 1972 and 1973, hearings were conducted on the performance of emergency core-cooling systems (ECCSs) under postulated loss-of-coolant-accident (LOCA) conditions. The hearings concluded with the Atomic Energy Commission (AEC) promulgating a rule governing the way ECCS performance would be calculated for a postulated LOCA. This rule, referred to as the "ECCS rule," and codified in 10 CFR 50.46 and in Appendix K to 10 CFR 50, was formulated from the viewpoint that very little applicable experimental data were available to substantiate ECCS performance predictions. As such, the rule contained a substantial number of requirements to treat certain thermal–hydraulic phenomena in a very conservative manner.

At the time the rule was promulgated, the AEC also charged the staff to conduct a research program to confirm the margins imposed in the ECCS rule. Responding to this direction, the staff embarked upon a long-term, broad-scope, thermal–hydraulic research program designed to produce a comprehensive data base from which the ECCS rule conservatisms could be qualified. This research program would become one of the premier research programs of the agency for the next 15 years. It comprised an extensive experimental program as well as a broad-based analytical code development program.[1] During approximately the first 10 years of this program, annual thermal–hydraulic research budgets were on the order of $60 million dollars.

As a result of the prominent position that this thermal–hydraulic research program held within the AEC, as well as in the industry and technical community, and along with its large budgets, many of the nation's top thermal–hydraulic experts were easily attracted to work on the program, either as contractors to the NRC or by joining the NRC staff. (This is, in fact, true with any large, high-visibility, well-funded program.) During this period the NRC operated several large-scale integral thermal–hydraulic facilities simultaneously.[2–4] These included Loss of Fluid Test (LOFT), Semiscale, and Multi-Loop Integral System Test (MIST). The NRC also engaged in cooperative test programs overseas, most notably the 2D/3D program with Germany and Japan, using the Upper Plenum Test Facility (UPTF), the Cylindrical Core Test Facility (CCTF), and the Slab Core Test Facility (SCTF).[5] In addition, a number of separate-effects experimental programs were being run at the same time. Code development efforts were also extensive.[6–8] The TRAC family of codes was developed along with continued development of RELAP.

In 1975, the NRC issued the *Reactor Safety Study*, commonly known as WASH-1400,[9] which was the first quantification of risk from nuclear plants using a systematic methodology known as probabilistic risk analysis (PRA). Although the study generated much controversy, it reached one conclusion that challenged the agency's wisdom regarding its perception of risk: the large-break LOCA (LBLOCA) was not the dominant source of risk to public health and safety.

For a number of reasons, however, this revelation did not produce immediate changes in the way the NRC expended its research dollars. One reason was that a substantial investment had already been made in test facilities and new code development that could not easily be changed. Another was that WASH-1400 was undergoing an intense, highly visible peer review, and thus people were reluctant to embrace its conclusions immediately. Finally, PRA was a relatively new "science" to the nuclear community, and many in the community were hesitant to embrace it. Therefore research efforts and most of the agency's research budget continued to focus on the LBLOCA.

The first LOFT test was run in 1978, and at that time the budget for conducting the LOFT program alone was about $40 million per year. (This amount was not solely for testing. It included funds for instrumentation development, code development, test analysis, and other indirect support activities.) The Semiscale program was running with a testing rate of about one test every $1^1/_2$ to 2 months. Semiscale had indirect costs similar to the LOFT indirect costs, costing the NRC about $7 million per year. With code development and other thermal–hydraulic programs, such as separate-effects tests, ongoing at the time, the total NRC budget for thermal–hydraulic research was about $60 million per year.

In March 1979 the Three Mile Island Nuclear Station Unit 2 (TMI-2) accident occurred. Until the pressurizer block valve was closed, this was in essence an unmitigated small-break LOCA (SBLOCA) initiated by a combination of equipment failures and operator errors. This accident helped to confirm the WASH-1400 conclusion that the LBLOCA was not the dominant contributor to risk. Following this accident, emphasis shifted fairly quickly from the study of the LBLOCA to other accidents of higher risk significance, in particular the SBLOCA and non-LOCA transients.

Although neither facility was specifically designed for SBLOCAs or non-LOCA transients, both LOFT and Semiscale were quickly modified to do SBLOCA and non-LOCA transient testing. However, by 1983, all necessary testing in LOFT was completed. The program was originally planned to terminate, but going from an annual budget of about $40 million to zero from one fiscal year to the next carried with it a major disruption of the Idaho National Engineering Laboratory (INEL), the national laboratory that ran LOFT. Such an abrupt closeout would put hundreds of technicians and engineers out of work if other programs could not be found for them to work on. To remedy this, the NRC, with the cooperation of the Department of Energy (DOE), proposed that the LOFT facility be used as an Organization for Economic Cooperation and Development (OECD) international consortium project. This project allowed the LOFT program to continue for 3 additional years at a funding rate of approximately $25 million per year. Thus INEL was provided a 3-year period in which LOFT personnel could be moved in an orderly manner to other programs.

With the completion of the NRC's LOFT program in 1983, the NRC's research budget began to decrease. In 1986, Semiscale completed all needed testing and was shut down. By 1988, the MIST facility, an integral thermal–hydraulic test facility that was part of a cooperative program among the NRC, Babcock and Wilcox (B&W), the Electric Power Research Institute (EPRI), and the B&W owners group to study SBLOCAs in the B&W reactor geometry, was completed. Also, in 1988, the NRC completed the revision to the ECCS rule, which now became a performance-based rule justified by 15 years of ECCS research.

By this time total NRC research budgets had dropped to about $100 million per year, and several other important research programs, in particular human factors, aging, and life extension, began to compete for the limited research dollars.

It was at this point, about 1989, that the NRC had to address the question of how to maintain a viable thermal–hydraulic research program within its budget limitations and with due consideration of other research priorities.

UNDERSTANDING THE NEED FOR THERMAL–HYDRAULIC RESEARCH

There are few who would argue that the thermal–hydraulic research program of the middle to late 1970s and early 1980s was not adequate. The top experts in the field were working on it, and research money was plentiful enough so that hardly anyone could complain that the issues of the day were not receiving enough attention. However, as with any program, as major issues are resolved and funding diminishes, fewer experts continue to work on the program.

This has been a source of concern for some who believe there has been a steady decline in the expertise available to the NRC in thermal–hydraulic research. It cannot be denied that some expertise has left. After 20 years, many of the pioneers in the thermal–hydraulic field have either retired, moved on to other technical areas, or still work in the field but in a different capacity. Many universities have closed their nuclear engineering departments, and thus the supply of bright, young nuclear reactor thermal–hydraulicists is substantially decreasing. In addition, there have been no new orders for nuclear plants since the 1970s, and most nuclear organizations have not grown in the past years. In the NRC's Office of Nuclear Regulatory Research, for example, there have been hardly any vacancies for new thermal–hydraulic engineers in the past several years. Therefore the measures used to judge the viability of thermal–hydraulic research today cannot be based on those of 10 to 15 years ago.

A new measure of success for a viable thermal–hydraulic research program is needed. This is a debatable and perhaps controversial subject, and it encompasses many unquantifiable considerations. Obviously, a general measure is whether the program meets the four key elements previously described; however, a more detailed evaluation is in order to determine the purpose of a thermal–hydraulic research program. The purpose varies, depending on the nature of the organization conducting the research.

A nuclear vendor is interested in both the safety of a nuclear reactor and its economic performance. Thus the accuracy to which a vendor must be able to predict steam-generator or core performance may be dictated by economics rather than safety.

In the 1970s, U.S. vendors developed codes to predict ECCS performance that used conservative models which were sufficient to comply with the ECCS rule but not unacceptably restrictive to plant performance. Once they developed these codes, development and experimentation continued only to the extent necessary to realize plant performance benefits. Further improvements solely for the quantification of safety margins were usually not forthcoming from the industry.

The NRC, on the other hand, has a charter for code development that is different from that of the vendors. The NRC charter is based only on safety considerations. Its objective is to develop codes with sufficient accuracy so that the staff can independently confirm the safety of licensed plants or those for which a license application is pending. However, this does not imply that industry codes can use simplified conservative models while NRC codes must use complex, best-estimate models that accurately represent all phenomena.

The NRC's basis for approving a design is the licensee's or applicant's analysis of its design (i.e., a computer code that adequately represents the plant and that has been acceptably validated against appropriate data must be used). Licensee or applicant codes are required to be compared with applicable experimental data, and it must be demonstrated that the codes are applicable to the designs that they are used to analyze.

The objective of licensee or applicant analysis is to demonstrate that the plant will not exceed established design limits for specific design-basis transients and accidents and will remain in a safe, coolable condition. Because the NRC allows these safety analyses to be performed using conservative assumptions, many times applicant or licensee codes will have built-in conservatisms that are acceptable for licensing but do not always allow the code to make accurate predictions of experimental data. Questions periodically come up regarding the analysis of an applicant or a licensee for which the staff cannot justify requiring the applicant or licensee to perform more experiments or analyses. Also, the staff is sometimes interested in safety margins associated with the design of an applicant or a licensee. The NRC codes are therefore used to perform confirmatory calculations of a vendor's design, including the assessment of design margins. They provide the staff with added assurance that the plant of the licensee or applicant will perform as expected. However, the NRC staff bases its licensing decisions on the code analyses of the applicant or licensee and not on the results of analyses using NRC codes.

Thus, although the NRC strives to develop the most accurate codes it can, in principle it should not have to meet criteria any more stringent than the strict safety criteria that the vendors' codes must meet. This premise cannot be overemphasized because it provides the basis for establishing limits on the NRC's thermal–hydraulic development program.

We have tried to establish the context in which the NRC's thermal–hydraulic research program must be developed and maintained today by examining the environment from which the program evolved and by describing the environment it exists in today. The next section describes how the NRC plans to develop and maintain a viable thermal–hydraulic research program within the current constraints of the NRC's resources, namely, a limited research budget that is more likely to shrink than grow in the coming years and no significant increases in NRC research staff.

There are also certain objectives that a thermal–hydraulic research program should strive to achieve:

1. Maintain a set of adequately validated codes with which NRC can perform confirmatory calculations.

2. Maintain a program of thermal–hydraulic research sufficient to attract and retain experts at NRC and among our national laboratories, private companies, and particularly universities that are available to interact and resolve nuclear reactor thermal–hydraulic matters.

3. Establish and resolve technical issues associated with the certification of new plant designs and the regulation of existing plants.

4. Provide state-of-the-art improvements, models, and codes for regulatory purposes so as to achieve an optimum capability in light of limited fiscal resources.

CODES

In 1987, the staff fully recognized that the budgets for thermal–hydraulic research were decreasing and began an intense effort to define the minimum level of support that would be needed to maintain a viable thermal–hydraulic research program. Our first task was to meet with the staff from the NRC's Office of Nuclear Reactor Regulation (NRR) to establish what their thermal–hydraulic analysis needs were. From these meetings, it was ultimately concluded that NRC would need, and therefore should maintain, four codes: TRAC-PWR at Los Alamos National Laboratory (LANL), TRAC-BWR at INEL,[a] RELAP5 at INEL, and RAMONA[10] at Brookhaven National Laboratory (BNL).

These codes were considered by NRR to be able to meet all their regulatory needs at that time. We then met with the managers of NRC thermal–hydraulic research programs at some of the national laboratories to discuss with them the level of effort (financial support) that would be necessary to retain a cadre of experts to work on the NRC codes. We concluded early on that a minimal level of code development work must be maintained because code experts were not likely to remain on an NRC program if meaningful, challenging work could not be provided. Thus, on the basis of our own experiences, the inputs we received from the NRC program managers at the national laboratories, and the current regulatory environment, we identified the four key elements for maintaining a robust, viable, thermal–hydraulic analysis capability for the NRC.

Challenging Technical Environment

As previously discussed, the best thermal–hydraulic experts were likely to leave the NRC's thermal–hydraulic program unless relevant, technically challenging work was provided. In 1987, no information was available on advanced designs sufficient to start work on them, but an accident-management program was initiated. Examining strategies for accident management fortunately involved thermal–hydraulic calculations, and thus some of the thermal–hydraulic experts at our national laboratories were able to work in that area as well.

In 1989, we began to receive sufficient information on the Westinghouse AP-600 and General Electric simplified boiling-water reactor (SBWR) designs to start examining our codes to see whether model improvements were needed to adequately model these designs. Both the NRC staff and the national laboratory managers agreed that advanced reactor thermal–hydraulic work would provide an outstanding challenge to the thermal–hydraulic experts at the laboratories.

In 1990, NRR was able to delineate its specific thermal–hydraulic needs for these advanced designs in detail. Most noteworthy was the need for a full-height, full-pressure thermal–hydraulic facility to simulate the AP-600 design and performance. Although it was preferable to design, construct, and operate such a facility in the United States, cost estimates were on the order of $50 million, an amount that was well beyond what NRC's existing research budget would permit. Most importantly, however, was that construction schedules showed that experimental data would not be available until after the scheduled AP-600 design certification, which would reduce its usefulness in the regulatory process.

It was finally decided to modify the ROSA (Rig of Safety Assessment) facility in Japan to conduct the AP-600 testing. National laboratory expertise in the United States was called upon extensively to analyze the proposed ROSA modifications necessary to properly simulate AP-600, analyze the expected AP-600 performance, and compare the results. The INEL did this work and also did the preliminary design work to establish the specifications for the ROSA facility modifications. The INEL will also be doing all the test predictions, posttest analyses, evaluations, code modifications, and model development and improvements that result from not only the ROSA testing but also the Westinghouse testing at the Simuluzione PWR per Esperienze di Sicurezza (SPES) facility in Italy and the research at Oregon State University. The INEL is also currently making all code modifications and model improvements necessary to the RELAP5 code for SBWR calculations. The LANL is developing a TRAC model of AP-600 and will be using it to analyze LBLOCA in the AP-600. Finally, BNL is using the RAMONA and RELAP5 codes to analyze the SBWR.

It is our belief that the extensive thermal–hydraulic code development programs at the national laboratories, as just described, are more than sufficient to maintain expertise in the field of thermal–hydraulics and to provide an exciting, technically challenging environment that will attract and retain some of the best experts in the field.

[a]The responsibility for maintenance of the TRAC/BF1 code was recently transferred from INEL to a private contractor, the Pennsylvania State University Nuclear Engineering Department.

Maintenance of a Cadre of Experts

For most of NRC's major codes, the scope of the code, the models within the code, and other areas such as numerics preclude any NRC person from being an expert on an entire code. These codes are, in fact, a team effort. On the basis of our discussions with the NRC program managers at the national laboratories and on our own previous experience, we conclude that, for each of the major codes (RELAP5 and TRAC), the minimum cadre of experts that should be held together as a team is about five. This, of course, varies with the extent of the use of the code. A smaller team would suffice for a code that is not extensively used. In this case, the cadre of experts would be maintained either through the use of student trainees or through code applications and assessment on another of the four codes.

The use of student trainees has both its benefits and limitations. The use of students in this type of effort provides excellent training in code development and fosters nuclear engineering education. A drawback is the inability of student-based groups to respond quickly in some cases when analyses are needed in a timely manner. With a staff-year of effort at a national laboratory costing an average of \$200 000, a rough estimate of a minimum funding level to maintain thermal–hydraulic expertise for the four codes previously mentioned would be on the order of \$4 million per year.

Any plan for maintenance of expertise must consider a proper mix of the unique assets available from national laboratories, universities, the unregulated private industry, and foreign governments and institutions. In the past, major experimental testing and code development in thermal–hydraulics has been at national laboratories. At present, the emphasis in testing has shifted to universities and foreign institutions. Major code development is still being accomplished at national laboratories, but code maintenance and improvement are shifting, where appropriate, to universities. On the other hand, code applications and code assessment are being accomplished among a variety of institution types.

Protection of proprietary data and the absence of conflicts of interest are important considerations in choosing institutions to perform research. Fortunately, the NRC has been able to obtain competent institutions and individuals to perform its research.

Assurance of Adequate Resources

One of the main difficulties associated with ongoing research programs is the acquisition of adequate funding. If funding fluctuates significantly from year to year, then it would be unreasonable to expect expert thermal–hydraulicists to remain on the program because not only would their continuous employment on the program be in constant jeopardy but also would the resources needed to carry out an effective program. Because the NRC's budget is established by the President and Congress, major budget changes (i.e., cuts) to some extent must be realized as cuts to individual programs. Nevertheless, we are adhering to a philosophy of stabilizing to the extent practical the thermal–hydraulic research budgets so that stable and predictable funding can be the expectation. Similarly, phaseouts or cancelation of programs should be done with sufficient advance warning to allow the individuals working on the program(s) in question to relocate to other programs in an orderly manner and thus preserve laboratory stability.

Continuing Involvement with Experimental Programs and the International Community

The goal of thermal–hydraulic research at the NRC is to produce an understanding of the thermal–hydraulic performance of commercial nuclear power plants so that their operation can be ensured without undue risk to the public health and safety. As stated earlier, this is being accomplished through the use of computer codes that have been validated against appropriate experimental data. Therefore establishment of a program associated with thermal–hydraulic experimental facilities should go hand in hand with an ongoing code development program. In accomplishing this goal, the NRC has revised its approach to experimental facilities from the approach followed in the 1970s and early to middle 1980s.

This was done for several reasons. First, there was no explicit need for large integral facilities following resolution of the LOCA issue. Second, as research budgets were reduced, the cost of running large, domestic, integral facilities became more prohibitive. Third, the staff concluded that smaller-scale, less-expensive facilities could, in fact, provide adequate code validation data.

The goal of the NRC's thermal–hydraulic research program is to maintain a strong involvement in ongoing experimental programs so that maximum experimental data can be made available for use in our code development program. We are actively reviewing and interacting with Westinghouse's AP-600 testing programs being conducted at the SPES facility in Italy and at Oregon State University. We continue to run the 1/9 linear-scale B&W simulation loop at the University of Maryland,[11] and we are sponsoring a testing program at the North Carolina State University[12] using their 1/9 linear-scale Freon loop, which simulates the Westinghouse Prairie Island Plant (2-loop).

We are sponsoring a major testing program on the AP-600 reactor in the 1/30 volume scaled ROSA V facility in Japan,[13] and we have recently awarded a contract to Purdue University for the construction of a small-scale integral loop that will simulate the General Electric SBWR.

A strong thermal–hydraulic research program must naturally take into account not only domestic programs but also international programs. In the area of thermal–hydraulic computer codes, the NRC enjoys a leadership role in the international community, primarily through its international Code Assessment and Maintenance Program (CAMP). Currently, more than 15 countries have requested and received NRC's thermal–hydraulic codes and are actively using them. In addition, more than 150 domestic organizations have requested and received our codes. A fundamental premise of receiving an NRC thermal–hydraulic code is to provide information back to the NRC on assessments performed, errors found, and model improvements that should be made or, in fact, were made. For that reason, NRC codes have been validated and verified more than any other codes in the world. The extensive international participation in CAMP attests to the continued international leadership enjoyed by the NRC's thermal–hydraulic research program.

Both NRC staff and our contractors also participate in international conferences on thermal–hydraulics and in international programs through organizations such as the International Atomic Energy Agency (IAEA) and the Committee on the Safety of Nuclear Installations (CSNI).

LONG-TERM PROJECTIONS

Both the AP-600 and SBWR are expected to be certified by 1995. Because of the finality aspects of the certification rule (10 CFR 52), it is expected that most of the research, in particular the thermal–hydraulic research, will be completed for these designs by that time. Another reactor, the Canadian Deuterium Uranium reactor (CANDU3), is expected to apply for certification in the 1995 time frame. This reactor has a different thermal–hydraulic design and performance compared with conventional U.S. designs. It is expected that, as thermal–hydraulic research on AP-600 and SBWR is completed, the research funds that become available will be applied to evaluating and assessing this design. The uniqueness of this design is expected to pose a strong technical challenge to the staff's thermal–hydraulic experts, both in-house and contractor, ensuring retention of expertise.

CONCLUSIONS

In the preceding sections, the question of how the NRC intends to maintain thermal–hydraulic expertise and a leadership role in the international community was addressed. The following are key elements of this approach:

- Provide a challenging technical environment.
- Maintain a cadre of experts.
- Ensure adequate resources.
- Continue involvement with experimental programs and with the international community.

By carrying out a thermal–hydraulic research program based on these four key elements, we believe that a healthy, viable, high-quality thermal–hydraulic analysis capability can be maintained by the NRC.

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Accident Analysis

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Contribution of the LOBI Project to LWR Safety Research

By C. Addabbo and A. Annunziato[a]

Abstract: *The Light-Water-Reactor Off-Normal Behavior Investigations (LOBI) Project has been carried out in the framework of the Commission of the European Communities Reactor Safety Research Program in close collaboration with institutional and industrial organizations of European Community member countries. The primary objective of the research program was the generation of an experimental data base for the assessment of the predictive capabilities of thermal–hydraulic system codes used in light-water-reactor safety analysis. Within this context, experiments have been conducted in an integral system test facility designed, constructed, and operated at the Ispra site of the Joint Research Centre. This article provides a brief historical perspective and summarizes major achievements of the research program, which is generally recognized as an effective approach to international collaboration in the field of reactor safety research.*

The Light-Water-Reactor (LWR) Off-Normal Behavior Investigations (LOBI)[1] Project has evolved over a time period characterized by a very intensive international effort in LWR safety research and development. In the early 1970s the level of understanding of thermal–hydraulic phenomenologies relevant to pressurized-water-reactor (PWR) postulated accident conditions, such as loss-of-coolant accidents (LOCAs) and the reliability of attendant computational methodologies, was rather primitive. A very limited experimental data base was available from integral system tests conducted in an initial, very crude configuration of the U.S. Nuclear Regulatory Commission (NRC) Semiscale Test Facility and early one-dimensional versions of such system codes as RELAP.

Since then a very large experimental data base has been gathered in several integral system test facilities, which, in addition to LOBI, include the Semiscale Program in the United States,[2,3] Loss-of-Fluid Test (LOFT) Program in the United States,[4,5] Primärkreisläufe (PKL) in Germany,[6] Large-Scale Test Facility (LSTF) in Japan,[7] BETHSY in France,[8] and Simuluzione PWR per Esperienze di Sicurezza in Italy;[9] also, the computational capabilities have matured to encompass a number of best-estimate codes such as RELAP, TRAC, ATHLET, and CATHARE. In the meantime, two major accidents occurred in the Three Mile Island Unit 2 (TMI-2) and Chernobyl power plants which had a significant impact on the reorientation of reactor safety research priorities from large-break LOCA safety issues to small-break LOCA and severe accident safety issues.

The Commission of the European Communities (EC) had been engaged in nuclear safety research activities since the signing in 1957 of the treaty establishing the European Atomic Energy Community (EURATOM). In line with its charter, the action of the Commission has been mainly devoted to the promotion of research and to the dissemination of results as well as to the establishment of uniform safety standards and practices among the member states. As required, the Commission promotes direct action research activities through the Joint Research Centre (JRC) and indirect action research

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activities in the laboratories of and in collaboration with research organizations of EC member states. Whenever it is practical and appropriate, the Commission encourages joint ventures with industrial or institutional organizations for the execution of analytical and/or experimental research activities in the JRC laboratories.

Within this context, the LOBI Project originated from a reactor safety research and development contract between the Commission and the Bundesminister für Forschung und Technologie (BMFT) of the Federal Republic of Germany in which, on the basis of contingent and perceived safety requirements, in 1972 the need for an experimental data base relevant to accident conditions in PWRs of German design was determined.

The contractual agreement, signed in 1973, envisaged the design, construction, and operation of an integral system test facility for the investigation of thermal–hydraulic phenomenologies pertinent to PWR large-break LOCAs; it was then renegotiated in 1982, extending the original research objectives to the investigation of phenomenologies pertinent to small-break LOCAs and to anticipated or abnormal transients, hereafter referred to as special transients. The BMFT program was then complemented by a Commission-specific program carried out in close collaboration with research organizations from EC member states that participated in the definition and specification of test cases reflecting national safety concerns.

RESEARCH RATIONALES

The LOBI research program, as initially conceived, has been mainly oriented toward the generation of an experimental data base relevant to postulated accidents and transients in PWRs. Specific research objectives included:

- Identification and/or verification of basic phenomenologies governing the thermal–hydraulic response of a scaled integral system test facility for a range of conditions relevant to LOCAs and special transients in PWRs.
- Generation of an experimental data base for the independent assessment of the predictive capabilities of large thermal–hydraulic system codes used in water reactor safety analysis.

The experimental program has been carried out in the LOBI test facility, a scale model of a four-loop PWR. The test facility was commissioned in December 1979 and operated until June 1982 in the MOD1 configuration for the investigation of large-break LOCAs; it was then extensively modified into the MOD2 configuration and was operated from April 1984 to June 1991 for the characterization of phenomenologies relevant to small-break LOCAs and special transients.

The executed experimental program, which in its final form includes 70 experiments, has been supported by comprehensive code application and assessment activities. ATHLET (DRUFAN), CATHARE, RELAP4, RELAP5, and TRAC have been largely used either within JRC or by outside organizations for test design and test prediction calculations. Development and application of advanced two-phase flow measurement techniques have constituted an integral part of the overall research strategy. A considerable effort has also been devoted to the development of an IBM version of the RELAP5 code,[10] which, together with various model improvements introduced at JRC, has been instrumental in enabling the calculation capabilities of many organizations within and outside the EC.

THE LOBI TEST FACILITY

The LOBI Test Facility[11] is a full-power, high-pressure integral system experimental installation representing an approximately 1:700 scale model of a four-loop, 1 300-MW(e) PWR. It incorporates the essential features of the reference reactor (Siemens-KWU Biblis B) primary and secondary cooling systems and is designed to preserve, within the general constraints of scaling criteria, prototypical system behavior under both normal and off-normal operating conditions.

System Configuration

The test facility comprises two primary loops, the intact loop and the broken loop, that represent, respectively, three loops and one loop of the reference PWR. Each primary loop contains a main coolant circulation pump (MCP) and an inverted U-tube-type steam generator (SG). The simulated core consists of an electrically heated 64-rod bundle arranged in an 8×8 square matrix inside the pressure-vessel model; nominal heating power is 5.3 MW(e). Each heater rod of the simulated core consists of an internally pressurized hollow tube with an active heated length of 3.9 m, an outer diameter of 10.75 mm, and a pitch of 14.3 mm. The wall thickness is varied in five steps to provide a cosine-shaped axial heat flux distribution. A lower plenum, an upper plenum, an annular downcomer, and an externally mounted upper head simulator are additional major components of the reactor model assembly. The primary cooling system, which is shown schematically in Fig. 1, operates at normal PWR conditions: approximately 15.8 MPa and 294 to 326 °C pressure and temperature, respectively. Heat is removed from the primary loops by the secondary cooling system, which contains a condenser and a cooler; the main feedwater pump; and the auxiliary feedwater system. Normal operating conditions of the secondary cooling system are 210 °C feedwater temperature and 6.45 MPa pressure. A summary of major test facility characteristics is given in Table 1.

The measurement system comprises about 700 measurement channels. It allows the measurement of all relevant thermohydraulic quantities at the boundaries (inlet and outlet) of each individual loop component and within the reactor pressure-vessel model and steam generators. A process control system allows the simulation of both main

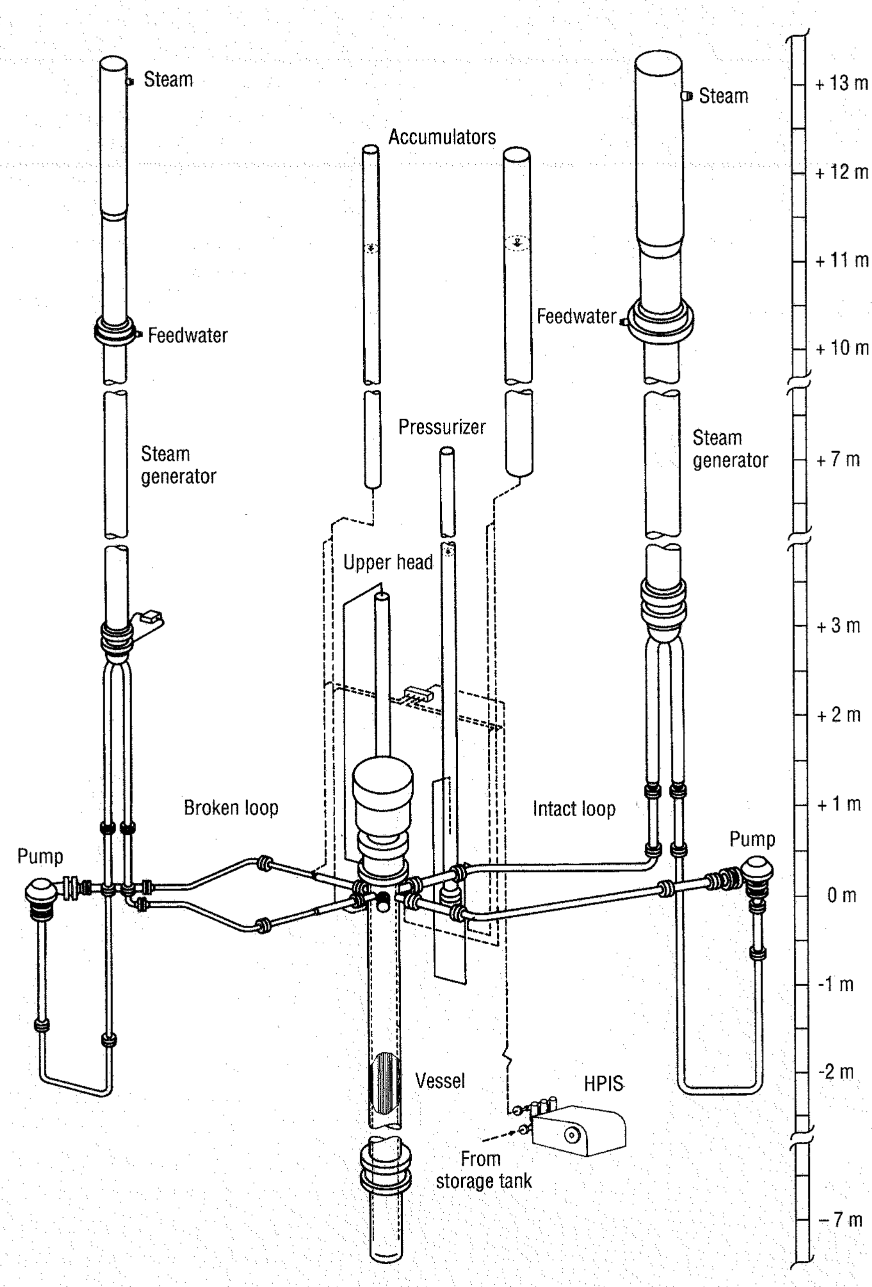

Fig. 1 Configuration of the LOB1-MOD2 test facility.

Table 1 LOBI Test Facility Characteristics

Primary system	Loops	2(1:3)
	Total volume	0.6 m^3
	Scaling factor	700
Core	Power	5.28 MW
	Length	3.9 m
	Number of rods	64
	Matrix	8×8
	Heater rods OD	10.75 mm
	Pitcher	14.3 mm
	Electrical heating	Direct
Downcomer	Configuration	Annular
	Gap width	50/12 mm MOD1 12 mm MOD2
Steam generators	Type	U-tubes
	Number of tubes	8 (broken loop) 24 (intact loop)
	Downcomer	Annular
Main coolant pumps	Type	Centrifugal
	Specific speed (DIN)	29.2
Nominal operation	Primary system	
	Pressure	15.8 MPa
	Temperature	294 to 326 °C
	Secondary system	
	Pressure	64 bar
	Temperature	210 to 280 °C
Organization	CEC Joint Research Centre	

coolant pump hydraulic behavior and core decay heat release. No provision has been made for the simulation of nuclear thermal–hydraulic feedback phenomena.

Scaling Rationales

The scaling rationales applied in the design of the test facility were aimed at preserving similarity of thermal–hydraulic behavior with respect to the reference plant. A power-to-volume scaling concept was adopted in the design of the facility to ensure the preservation of the primary and secondary fluid specific power.

The elevation of the major components was maintained at full height except for the pressurizer, which, while preserving the total volume and the steam-to-liquid volume ratio, was somewhat shortened to allow increased radial dimensions for the accommodation of the internal heaters. The core and the SG heat-transfer and flow areas were matched to the scale factor. Strict adherence to the power-to-volume scale factor would have resulted in unacceptably high wall frictional pressure losses in the primary loop pipework, which was thus appropriately shortened and increased in diameter to match the expected pressure drop in the reference plant.

In the MOD2 configuration of the test facility, special emphasis was given to the scaling of the steam generators because of their importance on the thermal–hydraulic evolution of small-break LOCAs and special transients. In particular, volume ratio, heat-transfer surface-to-volume ratio, hydraulic resistances, and elevations with specific respect to the lowest U-tube bend elevation were preserved.

A major exception to the general scaling concept is the design of the core vessel annular downcomer. The test facility has been configured with a downcomer of two different gap widths. Initially, a 50-mm downcomer gap was installed to mitigate bypass phenomena, which are largely influenced by excessive hot wall delay effects and countercurrent-flow limitations; this, however, resulted in a downcomer volume that was 6.3 times too large and, as a consequence, in an atypical system response during large-break LOCA experiments. The downcomer gap width was later changed to 12 mm, which was the result of a necessary technical compromise between the 7-mm scaled volume and the 25-mm pressure drop scaled width.

Simulation Constraints

With respect to the reference plant, the LOBI test facility, as any other scaled facility, has thus inherent distortions that may impair the typicality of the quantitative response of the installation. Although the height and the relative elevation of the major components have been preserved 1:1, the power-volume scaling concept adopted in the design of the installation has resulted in a configuration exhibiting a basically one-dimensional thermal–hydraulic response. The test results, therefore, cannot be directly extrapolated to assess the quantitative response of the full-size plant; rather, they provide a source of reference information for the understanding of basic thermal–hydraulic phenomenologies expected in PWR accident conditions and for the assessment of the predictive capabilities of system codes used in water reactor safety analysis.

THE LOBI RESEARCH PROGRAM

The LOBI research program comprises two experimental programs[12,13] with related analytical activities defined MOD1 and MOD2 to designate test facility configuration and specific research objectives:

- The MOD1 program was primarily conceived for the parametric investigation of large-break LOCA phenomenologies with main emphasis on emergency-core-cooling-system (ECCS) performance. It was conducted with the test facility in the MOD1 configuration from December 1979 to June 1982.

• The MOD2 program was carried out from April 1984 to June 1991 with the test facility in the MOD2 configuration. The relevant research priorities were directed toward the investigation of small-break LOCA and special transient phenomenologies with inclusion of recovery procedures and accident management strategies.

Both the MOD1 and the MOD2 experimental programs include two distinct test matrices, defined A and B.

• The test matrix A has been performed in the context of the contractual agreement with BMFT. The test cases were elaborated by German experts and LOBI staff members assembled in the Arbeitsgruppe LOBI-A (AG-LOBI-A), a subunit of the Sachverständigenkreis Notkühlung (SK-NK), a consultative committee to the BMFT of the German government.

• The test matrix B has instead been performed in the framework of the CEC Reactor Safety Research Program with independent contributions from several EC member countries. The test cases of this matrix were defined by experts from institutional and/or industrial research organizations of EC member countries assembled in the LOCA and Transients Program Task Forces, which are subunits of the LOBI Working Group B (WG-B), a JRC consultative body on the LOBI research program.

MOD1 Program

According to the special contractual agreements that originated the research program, the MOD1 phase of the LOBI experimental program was mainly devoted to the investigation of large-break LOCA phenomenologies. During the MOD1 testing phase, 25 LOCA tests covering the large-to-intermediate-break size range and 3 small-break LOCA scoping tests were successfully performed. Except for four tests that were executed in the framework of the Community program, most of the tests were specified by the German contract partner and were thus specified by delegated EC member country research organizations.

A summary overview of the LOBI-MOD1 experimental program is given in Table 2. With the exception of the initial 14 tests of the MOD1 program, which were performed with a large downcomer (50-mm gap width), all the other tests were performed with the small downcomer (12-mm gap width). Major parametric variations included break size, break location, downcomer width, main coolant pumps operation mode, and ECC injection mode.

MOD2 Program

The MOD2 phase of the LOBI experimental program reflected the change in emphasis in water reactor safety research that emerged in the early 1980s. A summary overview of the MOD2 program is given in Table 3. The overall test matrix included 42 tests, of which 16 were BMFT contractual, or A, tests and 26 were community, or B, tests. The range of postulated accidents included small-break LOCAs, special transients, and characterization tests; as appropriate, recovery procedures and accident management strategies were also investigated.

The primary objective of the MOD2 A tests was to complement and/or extend the existing experimental data base to postulated small-break LOCAs relevant to PWRs of Siemens-KWU design. Within this context, the MOD2 A tests were closely related to the MOD1 tests, which emphasized the large-break LOCA scenario, and to other experimental and/or analytical programs promoted by BMFT in the field of water reactor safety analysis. The B test cases reflected conditions of general interest for PWR safety analysis and included test conditions scaled to typical Westinghouse or Framatome PWR operating conditions.

The MOD2 test matrix contained 26 small-break LOCA tests covering a variety of initial and transient assumptions. Major characterizing features included break size, break locations, ECC injection mode, and MCP operation mode. The special transients test matrix included 12 test cases featuring primary system intact circuit faults and, as appropriate, plant recovery procedures and accident management strategies. Emphasis has been placed on loss of main feedwater, loss of main and auxiliary feedwater, station blackout, steam-line break, and feed-line break. Associated accident management procedures included primary and secondary feed and bleed and intentional primary system depressurization.

Test facility and component characterization tests have been an integral part of the research program; these tests include system heat loss measurements, secondary system inventory measurements, core bypass tests, SG performance tests, and natural circulation tests. The general objective of these tests was to characterize test facility atypicalities and basic heat transport mechanisms to provide data to reduce modeling uncertainties that could impair code predictive accuracy.

THE LOBI INTERNATIONAL CONTEXT

The international context in which the LOBI research program has been carried out has offered an opportunity

Table 2 LOBI-MOD1 Experimental Program (December 1979–June 1982)

Text	Sponsor	Date	Description[a]
		50-mm-Wide Downcomer Gap	
A1-04	Germany	12.12.79	200% CL Break LOCA CL ECC
A1-01	Germany	29.01.80	200% CL Break LOCA CM ECC
A1-02	Germany	14.02.80	200% CL Break LOCA CM ECC
A1-03	Germany	19.03.80	200% CL Break LOCA CM ECC
A1-04R	Germany	17.04.80	200% CL Break LOCA CL ECC
A1-05	Germany	06.05.80	200% CL Break LOCA CM ECC
SD-SL-01	Germany	04.06.80	10% CL Break LOCA CL ECC
SD-SL-02	Germany	18.06.80	1% CL Break LOCA CL ECC
SD-SL-03	Germany	24.09.80	0.4% CL Break LOCA CL ECC
A2-59	Germany	27.10.80	100% CL Break LOCA CL ECC
B-101	France	26.11.80	2 × 50% CL Break LOCA CL ECC
A2-55	Germany	19.01.81	50% CL Break LOCA CL ECC
A2-59R	Germany	11.02.81	100% CL Break LOCA CL ECC
B-R1M	Germany	17.03.81	25% CL Break LOCA CL ECC
		12-mm-Wide Downcomer Gap	
A1-66	Germany	03.07.81	200% CL Break LOCA CL ECC
A1-07	Germany	09.07.81	200% CL Break LOCA no ECC
A1-06	Germany	21.07.81	200% CL Break LOCA CM ECC
A1-67	Germany	30.09.81	25% CL Break LOCA CM ECC
A1-68	Germany	28.10.81	50% CL Break LOCA CM ECC
A1-10A	Germany	25.11.81	200% HL Break LOCA CM ECC
A1-10B	Germany	10.12.81	200% HL Break LOCA CM ECC
A1-70	Germany	13.01.82	200% PS Break LOCA CM ECC
A1-73	Germany	04.02.82	25% HL Break LOCA CM ECC
A1-72	Germany	24.03.82	200% CL Break LOCA CM ECC
A1-69	Germany	06.04.82	100% CL Break LOCA CM ECC
A1-74	Germany	21.04.82	200% CL Break LOCA CM ECC
B-222	France	05.05.82	2 × 50% CL Break LOCA CL ECC
B-302	Italy	16.06.82	2 × 50% HL Break LOCA CL ECC

[a]Abbreviations:

CL	Cold leg	HL	Hot leg
CM	Combined model hot + cold leg	LOCA	Loss-of-coolant accident
ECC	Emergency core cooling	PS	Pump suction

for a close collaboration among delegates of national research laboratories. It has also provided an independent forum for the exchange of expertise and information on reactor safety-related matters.

Test Allocation

As previously mentioned, although the tests of the A matrix were exclusively defined by the German contractual partner, the tests of the B matrix were allocated to EC member countries through representing research organizations that, on the basis of specific interests, took charge of the responsibility to collaborate with the LOBI staff in the detailed specification of the test profile as well as in the pretest and posttest analysis of the results.

Counterpart Tests

Large system codes used in reactor safety analysis are generally benchmarked against experimental data from scaled integral system or separate effects test facilities. Comparison of the predicted transient response with test data from the full-size plant would be desirable, but this is clearly prohibitive for obvious economic and practical considerations; controversy often arises when the predictive capability of a system code is scaled up. To this end it is therefore desirable to assess the code against a set of data obtained from different scale test facilities.

Within this context, a few tests of the MOD1 and MOD2 experimental programs were defined and executed as counterpart to similar tests performed in other test facilities, such as Semiscale, PKL, BETHSY, LSTF,

Table 3 LOBI-MOD2 Experimental Program (April 1984–June 1991)

Test	Sponsor	Date	Description[a]
A1-76	Germany	12.04.84	SG Performance
A2-81	Germany	27.09.84	1% CL break LOCA, 2/4 HPIS in CL
A1-82	Germany	28.09.84	1% CL break LOCA, 2/4 HPIS in HL
A1-78	Germany	24.10.84	2% CL break LOCA, ECC in CM
A2-77A	Germany	28.11.84	Natural circulation, 90 and 75 bar
A1-83	Germany	19.12.84	10% CL break LOCA, ECC in CM
A2-90	Germany	27.03.85	LONOP-AATWS "Station Blackout"
A1-85	Germany	07.05.85	0.4% Pressurizer break, 2/4 HPIS in HL
BL-00	France	03.07.85	0.4% CL break LOCA, 1/3 HPIS in CL
A1-84	Germany	14.10.85	10% HL break LOCA, ECC in CM
BT-00	Great Britain	30.11.85	LOFW + feed and bleed
BT-01	Belgium	24.01.86	Small (10%) steam line break + PTS
BL-02	Great Britain	22.03.86	3% CL break LOCA, 2/4 HPIS in CL
A1-79	Germany	15.05.86	1% CL break LOCA, 4/4 HPIS in HL
A1-88	Germany	11.06.86	0.4% CL break LOCA, as cooldown
BL-01	Germany	20.09.86	5% CL break LOCA, HPIS + ACCU in CM
BC-01	WG-B	18.10.86	SG secondary inventory
BC-02	WG-B	26.11.86	SG heat losses
BL-21	Italy	24.01.87	0.4% SGTR + "SSN" recovery
BL-12	France	19.02.87	1% CL break LOCA, no HPIS, no cooldown
BT-02	France	09.05.87	LOAF + feed and bleed
BT-12	Great Britain	17.06.87	Large (100%) steam line break
A1-91	Germany	26.09.87	1% CL break LOCA, 1/4 HPIS in HL
BT-03	Italy	24.10.87	LOFW-ATWS + "SSN" recovery
A1-92	Germany	30.11.87	Natural circulation, 40 bar
BL-16	Germany	19.03.88	0.4% CL break LOCA, as cooldown
BC-03	WG-B	15.04.88	SG heat losses
A1-93	Germany	30.04.88	2% CL break LOCA, no HPIS
A1-94	Germany	27.05.88	4% CL break LOCA, 40 bar
BC-04	WG-B	15.04.89	5% CL break LOCA, HPIS + ACCU in CL
BL-22	Belgium	17.06.89	0.4% SGTR + cooldown
A1-87	Germany	11.11.89	PCS cooldown, MCP off
BT-04	France	10.02.90	PCS cooldown, MCP on, 1-SG isolated
BL-34	WG-B	22.03.90	6% CL break LOCA at low power, BETHSY CPT
BL-44	JRC	26.04.90	6% CL break LOCA at full power, no HPIS
BT-56	Great Britain	03.07.90	Multiple failures
BT-15/16	Great Britain	22.11.90	LOFW with SG boiloff and refill, MCPs on/off
BT-17	Germany	07.02.91	LOFW with SG feed and bleed
BT-06	France	21.03.91	Small (10%) feed line break
BL-40	Spain	16.05.91	SGTR in 1-loop PWR
BL-06	France–Great Britain	21.06.91	1% CL break LOCA, HPIS off, MCP on

[a] Abbreviations:

ACCU	Accumulator	LOFW	Loss of feedwater
ATWS	Anticipated transient without scram	LONOP	Loss of on-site and off-site power
BETHSY	French test facility	MCP	Main coolant pump
CL	Cold leg	PCS	Primary cooling system
CM	Combined model hot + cold leg	PS	Pump suction
ECC	Emergency core cooling	PTS	Pressurized thermal shock
HL	Hot leg	SG	Steam generator
HPIS	High-pressure injection system	SGTR	Steam generator tube rupture
LOAF	Loss of all feedwater	SSN	Core rescue system
LOCA	Loss-of-coolant accident		

and SPES.[14–16] These tests were performed under similar initial and transient assumptions and thus provided a set of data that emphasized the relevance of geometrical scaling parameters on the qualitative rather than on the quantitative evolution of the prospective test case.

International Prediction Exercise

The very first large-break LOCA test of the LOBI-MOD1 experimental program, test A1-04, was used for a special type of blind standard problem exercise, the LOBI

Pre-Prediction Exercise (PREX); 16 participants from various EC member states and the United States submitted calculations using a number of system codes.

The first small-break LOCA test of the LOBI-MOD2 experimental program, test A2-81, was designated by the Committee on the Safety of Nuclear Installations (CSNI) of the Organization for Economic Cooperation and Development (OECD) as International Standard Problem 18 (ISP-18); 27 participants from European and North American organizations provided prediction calculations with 12 codes or code versions.[17]

OUTLINE OF LOBI EXPERIMENTAL PROFILES

The experimental profiles of significant test cases covering typical accident conditions of interest to the safety analysis of PWRs are outlined in the following sections. Generally, the methodology used in the definition of each test case and in the establishment of the corresponding test profile was to reproduce governing physical phenomena rather than plant-specific behavior.

Large-Break LOCAs

Twenty-five tests addressing phenomenologies relevant to postulated design-basis LOCAs in PWRs have been performed with the test facility in the MOD1 configuration. Primary emphasis was placed on the performance of the accumulator safety injection system (ACCU) during the early blowdown phase of a LOCA; in the MOD1 configuration of the test facility, the high-pressure injection system (HPIS) and the low-pressure injection system (LPIS) were not represented.

Typically, in a double-ended (200%) cold-leg-break LOCA simulation, the initial blowdown phase is characterized by a large discharge of subcooled water from the break and a fast depressurization of the primary cooling system; thereafter the depressurization rate decreases as saturated critical flow is established at the break orifice.

The core thermal response is characterized by an early departure from nucleate boiling (DNB). The initial heater rod temperature rise then decreases or even reverses (early rewet) because of the combined effects of reestablished core flow and core power decay resulting from negative reactivity feedback. Eventually the loss of forced circulation and wide voiding cause the core to undergo a new steady temperature rise until ECCS is effective.

The initial large-break LOCA tests were conducted with the test facility configured with a largely overscaled downcomer gap of 50 mm to minimize downcomer countercurrent flow limitation as well as hot wall delay phenomena, especially during the refill period. Thereafter the downcomer gap was reduced to 12 mm to ensure a correct volume-scaled fluid distribution within the primary cooling system.

The influence of downcomer gap width and hence pressure-vessel volume on overall system response during a large-break LOCA has no immediate relevance for the safety evaluation of current PWRs if the downcomer dimensions are fixed; however, it may have some conceptual relevance for the design of new plants. The relatively higher liquid inventory initially available within the larger 50-mm downcomer ensured better core cooling and a more pronounced post-DNB early rewet that extended over the entire heated length of the simulated core (Figs. 2 and 3). Early rewet during blowdown has also been observed in the LOFT large-break LOCA experiments; this occurrence supports the characterization of this phenomenon as thermal–hydraulic controlled and also significantly influenced by MCP operation mode.[18–20]

For PWRs presently in operation, ECC water injection is performed, according to vendor type, at different locations [either only into the cold leg (Westinghouse and Framatome) or combined into both cold and hot legs (Siemens-KWU)]. The influence of these different injection modes was investigated with the LOBI MOD1 test facility configured with the 12-mm downcomer. Because of the nearly one-dimensional characteristics of the test facility, no conclusive statement can be made with respect to the relative effectiveness of the different ECC injection modes. The penetration of ECC water injected in the hot leg was somewhat limited by countercurrent flow limitation at the upper tie plate; this phenomenon is certainly less severe in the full-size plant where it is suppressed by flow channeling phenomena.

Small-Break LOCAs

Small-break LOCA tests covering a wide range of parametric variations have been performed with the test facility in both the MOD1 and MOD2 configurations. Although the specific objectives of the three small-break LOCA tests performed with the MOD1 configuration were scoping in nature, early useful information on natural-circulation energy transport mechanisms was nevertheless obtained. With the LOBI facility in the MOD2 configuration, 23 small-break LOCA tests have been performed; 11 in the framework of the BMFT program and 12 in the framework of the Community program. The range of break size varied from 0.4 up to 10%; cold leg, hot leg, pressurizer top, and SG U-tube ruptures

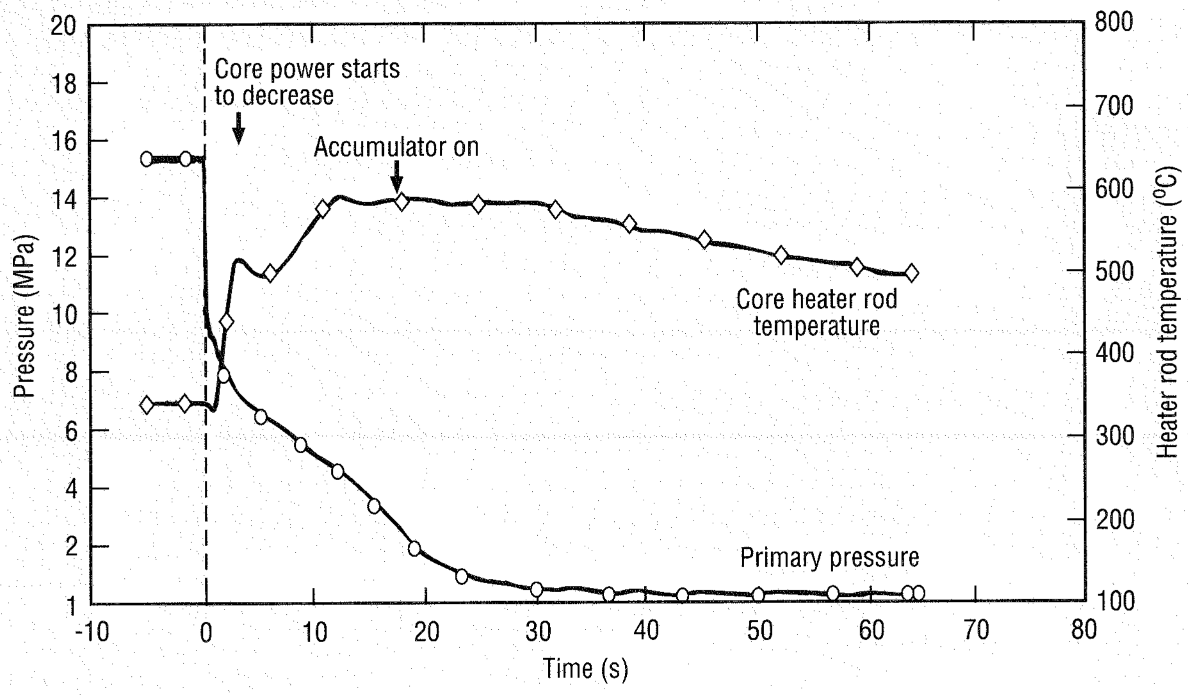

Fig. 2 Large (200%) cold-leg-break loss-of-coolant accident with the small (12-mm) downcomer, primary system pressure, and core temperature responses.

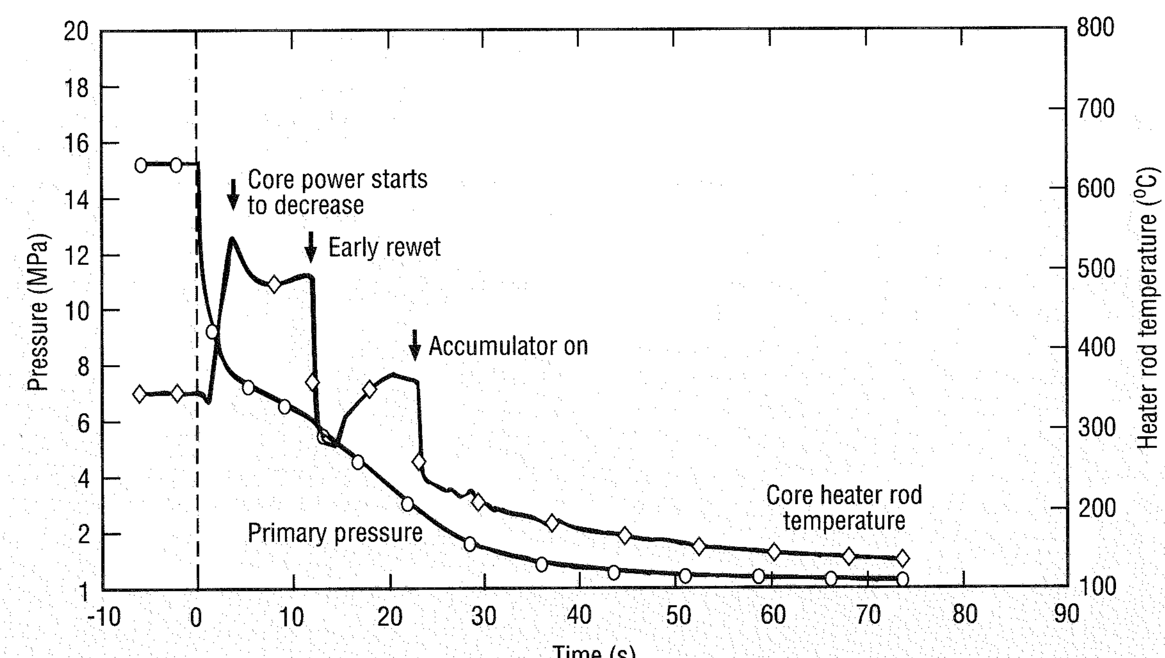

Fig. 3 Large (200%) cold-leg-break loss-of-coolant accident with the large (50-mm) downcomer, primary system pressure, and core temperature responses.

were simulated. The ECC was generally aligned with the hot leg or in the combined injection mode for the A tests and with the cold leg for the B tests. The secondary cooling system, when not otherwise specified, was automatically cooled down at a preset rate. Initial test conditions were generally similar.

The primary system response to a small-break LOCA is initially characterized by a relatively slow depressurization coupled with a progressive voiding starting from the upper elevations. The voiding process is eventually reduced or even arrested by fluid makeup from the HPIS and ACCU ECC systems. With the ECCS operational, heater rod temperature excursions were observed in the 10% break-size test case (Figs. 4 and 5). Here the first core dryout/rewet sequence was governed by loop seal formation and clearout; the second sequence was governed by core boiloff and inventory replenishment by ECC injection from the ACCU. A tendency to loop seal formation in both the intact and broken-loop crossover legs was observed with decreasing break size; definite clearout was observed in the 10% (both loops) and in the 5% (intact loop only) test cases. Liquid holdup in the hot-legs pipework was seen to be influential on loop seal behavior.

In the event of small-break LOCAs in PWRs, the secondary system is generally used as an additional heat sink to guarantee adequate primary system energy removal and depressurization. In the LOBI-MOD2 test facility and under the relevant test conditions, the primary system followed closely the secondary system cooldown for the range of break sizes 1% and smaller; for break sizes 2% and higher, primary and secondary system cooldown decoupled with the break flow is sufficient to ensure effective primary system energy removal and depressurization.[21]

The influence of HPIS capacity on overall system behavior and, in particular, on core thermal response was investigated for the 1% cold-leg-break LOCA configuration with the HPIS injection rate varied from full (4/4 trains available) to zero capacity. Core coolability was generally ensured with the HPIS operational; severe heater rod temperature excursion was observed with the HPIS disabled. The influence of MCP operation mode was investigated in a 1% cold-leg-break LOCA without HPIS injection; test results indicate that under these conditions the difference in overall system response is not significant.

The HPIS is directed in certain plants into the hot rather than into the cold legs to mitigate or even avoid potential pressurized-thermal-shock loads on the pressure-vessel shroud. This arrangement would prevent subcooled ECC water from reaching the downcomer directly while providing effective core cooling. This was clearly confirmed by a comparative analysis of the results from two 1% cold-leg-break LOCA tests that were identical in both initial and boundary conditions with the exception of the HPIS alignment.[22]

Steam generator tube rupture (SGTR) phenomenologies were investigated under typical emergency operating procedures, such as (1) an initial phase without operation intervention, (2) a consolidation phase to attain an adequate primary system subcooling, and (3) a secondary-side controlled depressurization phase (Fig. 6). Alternative recovery procedures in the event of HPIS unavailability or specific phenomena in a one-loop plant configuration were also investigated.[23]

Special Transients

The performed anticipated and/or abnormal transients were prioritized, taking into account system codes development and assessment purposes; as appropriate, phenomenologies relevant to prospected recovery and accident management procedures were investigated. The performed tests comprise station blackout, loss of feedwater, steam-line break, and feed-line break occurrences with relevant recovery procedures. Typical test profiles are reported in Figs. 7 and 8.

A loss of off-site and normal on-site electrical power anticipated transient without scram (LONOP-ATWS), which could be referred to as "station blackout" if auxiliary diesel power is considered, was performed on request of the German contract partner. To stay below the maximum operating pressure of the test facility (17.0 MPa) and to allow a representative pressure and temperature increase following the inception of the simulated fault, the pretransient steady-state pressure was set at 14.0 MPa (instead of 15.8 MPa); accordingly, the secondary system pressure also was reduced to 5.0 MPa (from 6.45 MPa). The primary and secondary safety valve set points were 15.2 MPa and 7.9 MPa, respectively. The initial transient then evolved through the SG boiloff and refill phases that, however, are not to be necessarily related to the progression of the initiating fault (Fig. 7).

Five loss-of-feedwater tests have been performed within the Community program framework. The first test represented a loss of main feedwater defined by the former UK Central Electricity Generating Board (CEGB), now Nuclear Electric (NE), with the Sizewell-B as reference reactor system. The initial fault was then followed by the simulation of loss of emergency feed and by a recovery phase featuring primary system feed and bleed. The second test was defined by the French Commissariat

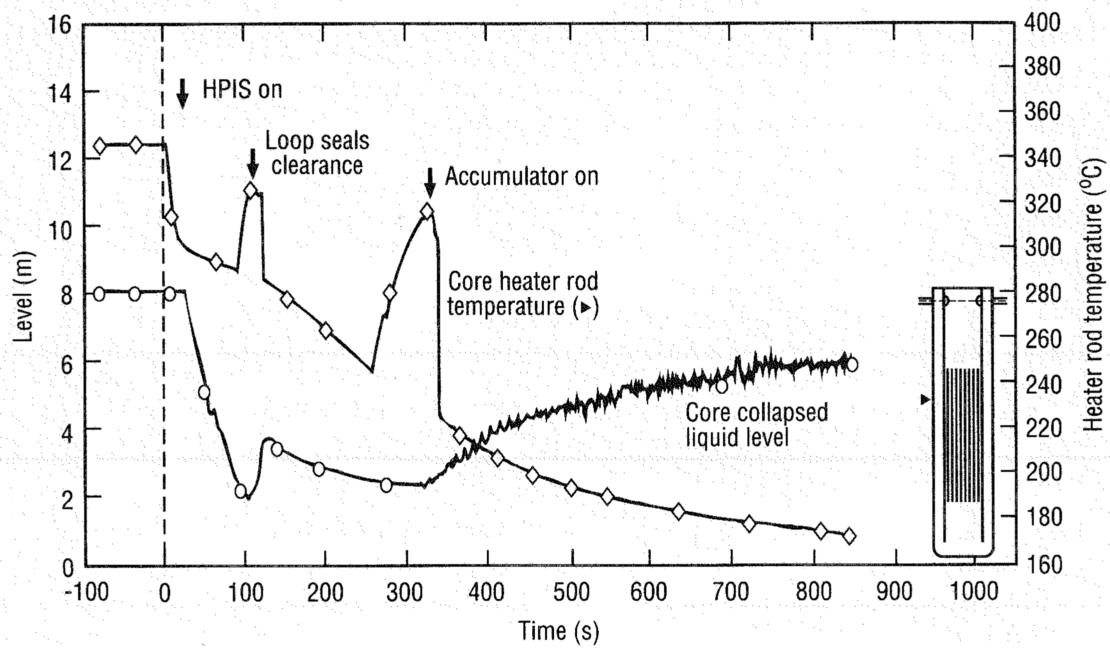

Fig. 4 Small (10%) cold-leg-break loss-of-coolant accident with core temperature and collapsed liquid-level responses.

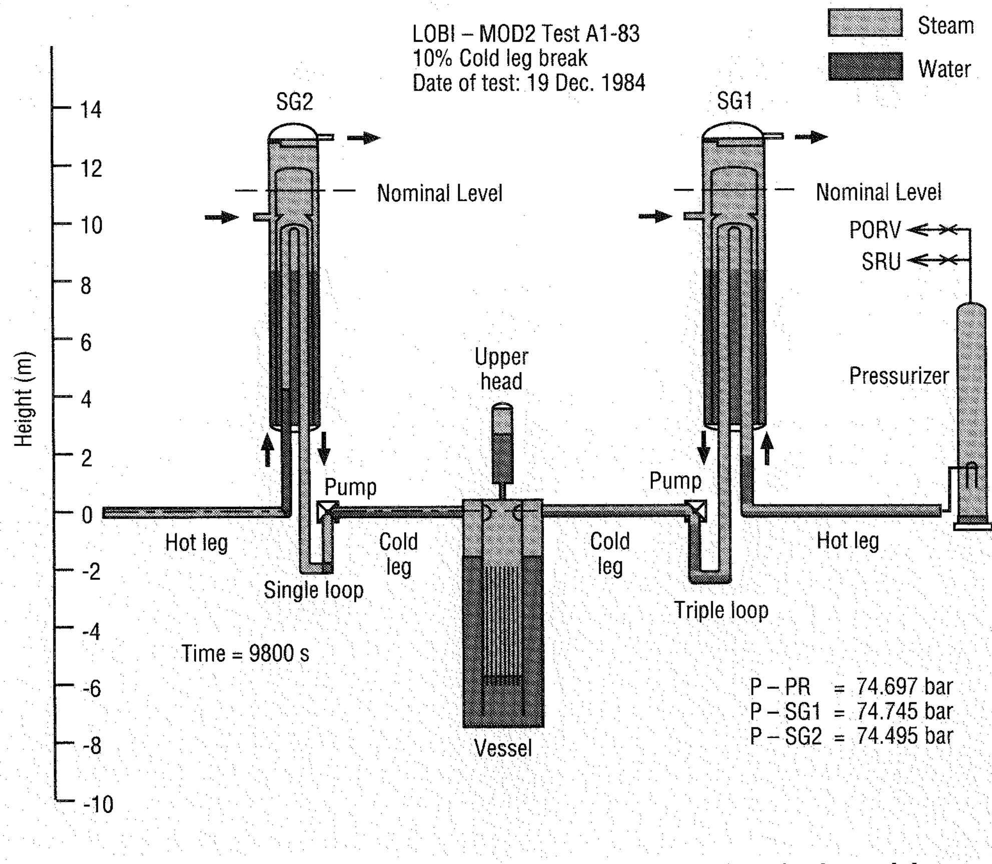

Fig. 5 Small (10%) cold-leg-break loss-of-coolant accident with fluid distribution before loop seal clearance.

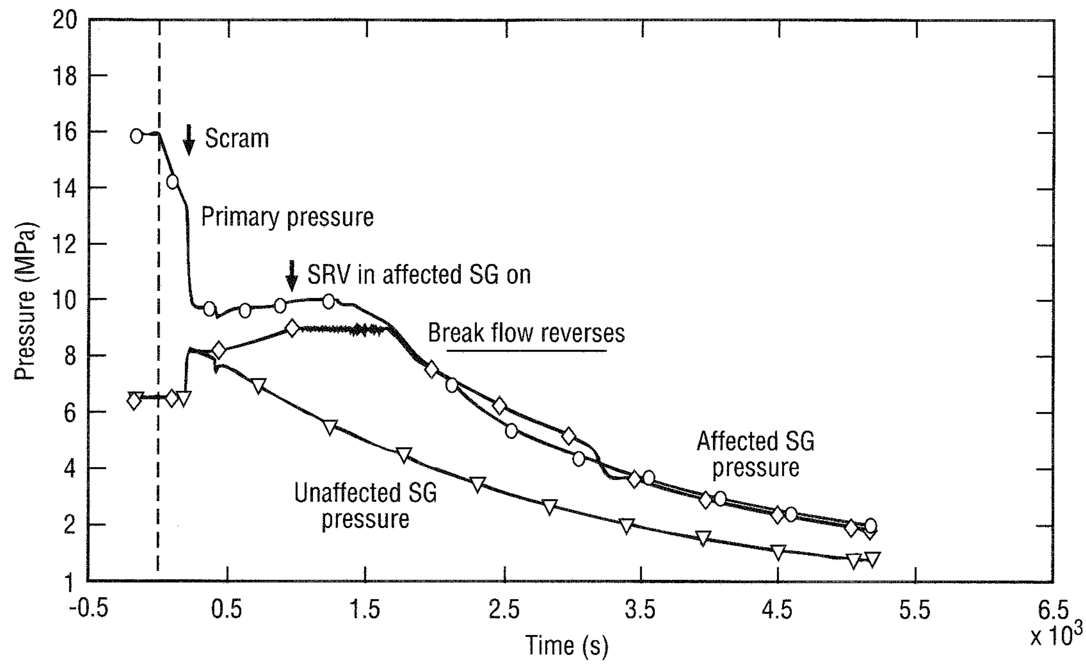

Fig. 6 Small (0.4%) steam generator tube rupture with primary and secondary system pressure responses.

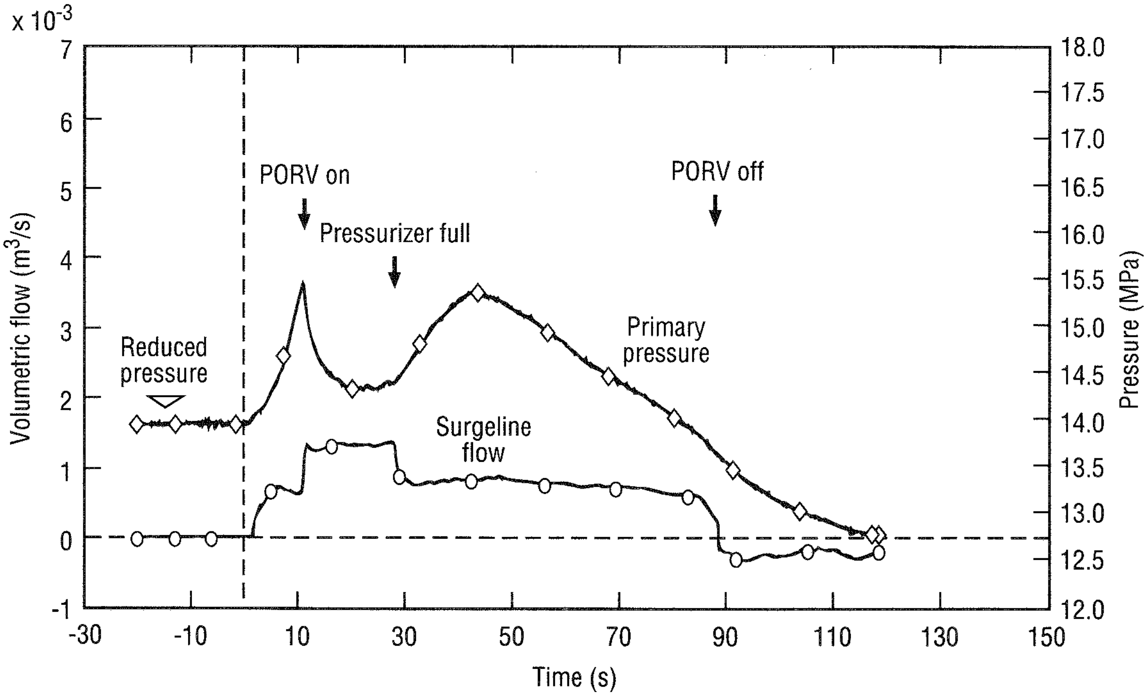

Fig. 7 Station blackout, surge-line volumetric flow, and primary system pressure responses.

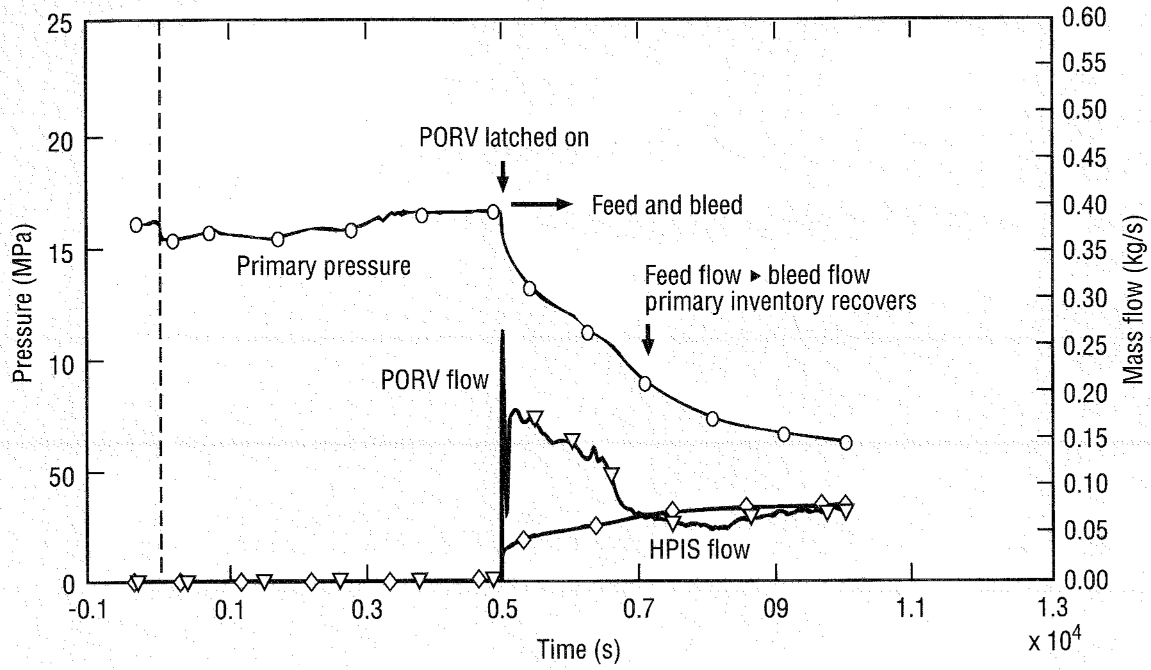

Fig. 8 Loss of feedwater with primary system feed and bleed, primary system pressure with power-operated relief valve, and high-pressure injection system flow rates.

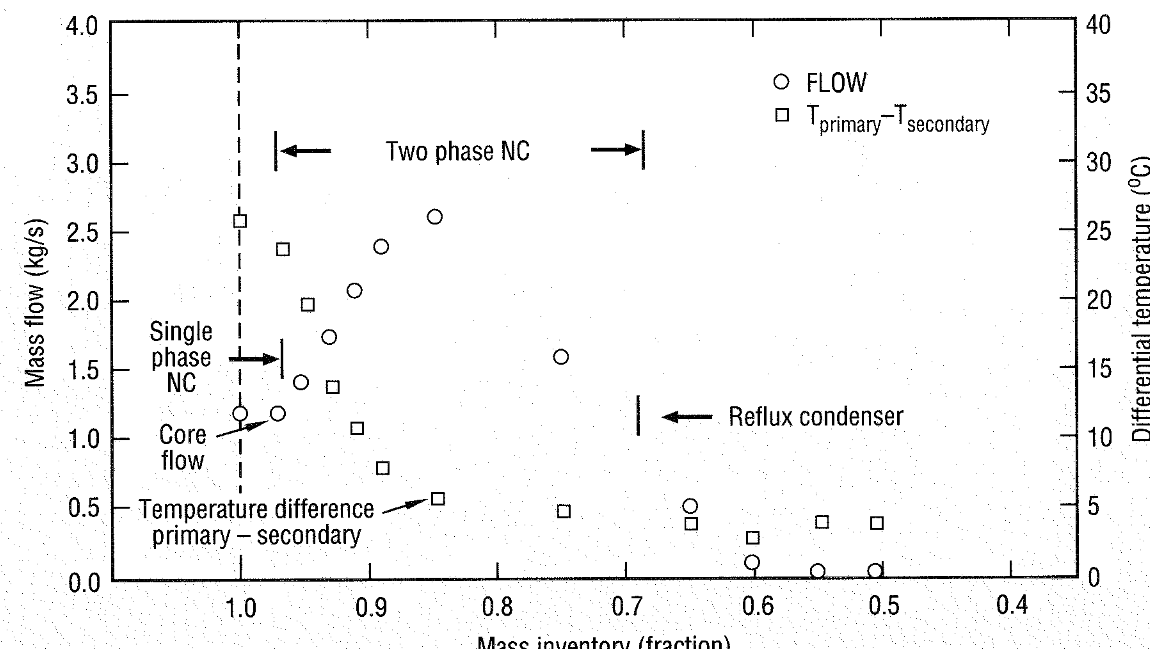

Fig. 9 Natural circulation at 4.0 MPa, core flow rate, and primary-to-secondary temperature difference vs. system mass inventory.

a l'Energie Atomique (CEA) and represented a loss of all (main and emergency) feedwater, which was then eventually terminated also by primary system feed and bleed (Fig. 8). The third test of the series represented an ATWS test case originally defined by the Italian Committee on Nuclear and Alternative Energies (ENEA); this test was terminated by a passive recovery procedure featuring intentional primary system depressurization through an equivalent 6-in. valve on top of the pressurizer and feed through ACCU at a set pressure of 4.2 MPa.

In reference to PWR secondary system faults, steam-line and feed-line break transients are generally analyzed to verify the mitigative features of the engineered safety systems with respect to both plant integrity and environment protection. Small (10%) and large (100%) steam-line break tests have been simulated on request from the UK-NE (CEGB) and the Belgian TRACTEBEL; a small (10%) feed-line break proposed by the French CEA has also been performed. In reference to the steam-line break test cases, water carryover in the steam line was negligible in the small-break test case and significantly low in the large-break test case, which indicates effective separator efficiency.

Natural-circulation heat-transport mechanisms have been investigated in two tests defined by the German contract partner. The first test, which was performed at a prevailing primary system pressure of 9.0 MPa, exhibited strong flow oscillations during the transition from two-phase natural circulation to reflux condenser heat transport.[24–26] The second test was performed in the framework of the LOBI-PKL counterpart test program and was thus performed at a prevailing primary system pressure of 4.0 MPa (Fig. 9).

Accident Management

In addition to primary system "feed and bleed,"[27] secondary system "feed and bleed"[28,29] is also being considered as an accident management procedure for the mitigation of the consequences of intact circuit faults. This procedure was investigated in a transient initiated by loss of main and emergency feedwater, which revealed a strong dependence on primary and secondary system condensation and evaporation processes. From comparative analyses of the LOBI results with the results of a similar test performed in the SPES test facility,[30] it can be inferred that one steam generator could be sufficient in ensuring an effective recovery procedure.

For the mitigation of the consequences of a small LOCA in the event of loss of safety injection, pressurizer-relief valves can be used to envisage the enhancement of the ECC safety injection system through the intentional, operator-managed increase of primary system depressurization. The event sequence pertinent to such an accident management procedure was verified in an SGTR test case, in an LOFW test, and in a 2%

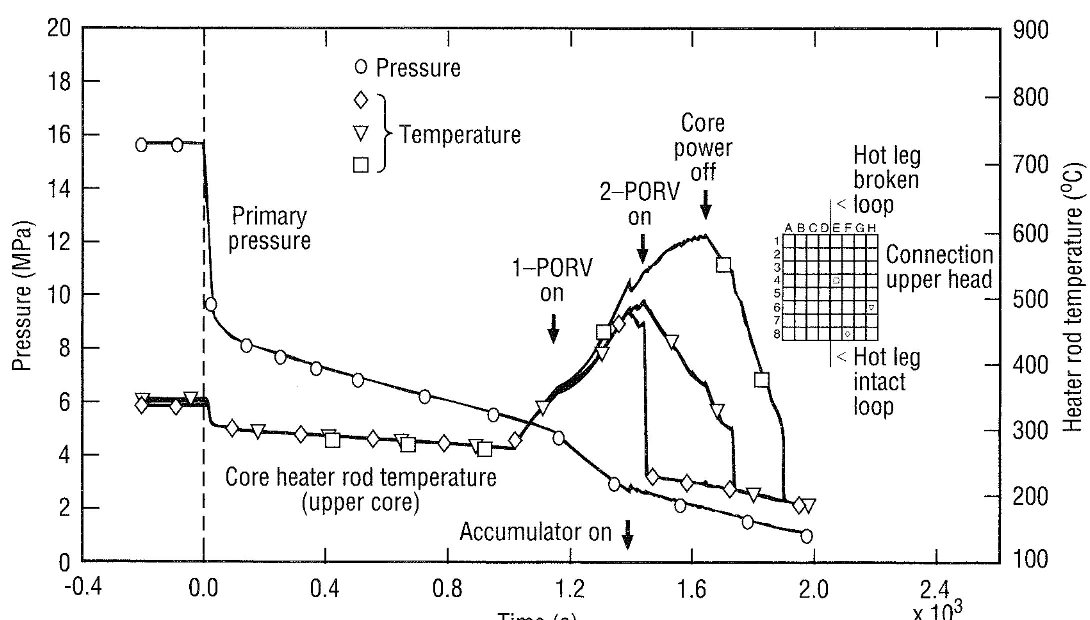

Fig. 10 Small (2%) cold-leg-break loss-of-coolant accident with intentional primary system depressurization, primary system pressure, and upper core temperature responses.

cold-leg-break LOCA (Fig. 10). In reference to the last, following core dryout as the HPIS was disabled, two pressurizer-relief valves were latched open in sequence to enhance primary depressurization. This caused the earlier intervention of the ACCU and holdup of heater rod temperature rise or even localized rewetting at the uppermost elevations.

CONCLUSIONS

The LOBI Project has provided a substantial contribution to the overall international effort dedicated in the last two decades to reactor thermal–hydraulic safety research. A comprehensive data base consisting of 70 experiments spanning a wide range of thermal–hydraulic phenomenologies expected in PWR accident conditions and of direct relevance for the assessment of system codes used in water reactor safety analysis has been acquired.

As structured, the LOBI Project has represented an effective approach to international collaboration in the field of reactor safety research and development. The international framework in which it has been carried out has also provided an independent forum for a systematic exchange of technical and scientific information among national experts and an opportunity to strive for a consensus of opinions on criteria and methodologies adopted in the safety analysis of water-cooled reactors.

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Aluminum–Uranium Fuel-Melt Behavior During Severe Nuclear Reactor Accidents

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Abstract: *The behavior of aluminum–uranium alloy nuclear reactor fuels in severe melting accidents is assessed. The results from several in-pile overheating incidents and from several experimental tests are used to derive conclusions regarding melt behavior and fission-product release in severe reactor accidents. These assessments indicate three distinct stages of fuel failure, which are described in detail. Experimental results that illustrate the foaming behavior of irradiated metallic fuels are also presented. These data describe the importance of the oxide film on the surface of the molten fuel in determining the fuel relocation behavior. The foaming and swelling of the fuel also are shown to correlate with the fission-product release phenomena.*

Tubular or plate-type fuels composed of an aluminum–uranium (Al–U) core and aluminum cladding have been widely used in research reactors and in the Savannah River Isotopic Production Reactors. Similar fuels with aluminum cladding and a uranium oxide (U_3O_8) dispersed particulate have also been used in several reactors. Both compositions make possible a high neutron flux and have the advantage of excellent heat transfer properties and relative ease of fabrication and reprocessing.

The Al–U metallic fuels are relatively low melting. The alloy fuels typically melt at about 650 °C. This low melting temperature is only a concern when an abnormal condition interferes with cooling. The result of such a condition could be localized fuel damage or even extensive melting of the reactor core.

The experimental results presented in this article show the foaming and failure behavior of irradiated Al–U alloys during melting. These data show the importance of the oxide film on the surface of the melt in determining its relocation mechanics. The assessments presented indicate three distinct stages of fuel failure at heating rates of less than 100 °C/s. At temperatures about 100 °C below the melting point of aluminum, blistering of the aluminum cladding is observed. At slightly higher temperatures, the cladding fails by cracking. When the fuel melts, it can flow through the gaps in the cladding. Low-burnup metallic fuel flows as rivulets over the surface of the oxidized cladding; for high-burnup metallic fuels, a molten metallic foam is exuded.

The data and observations presented are derived from two sources: laboratory studies of fuel melting under controlled conditions and incidents that have occurred during the operation of the Al–U fueled reactors. The latter include in-pile melting tests. Table 1 gives a list of these sources correlated with the phenomena observed.

The data sources and results from Table 1 are described in greater detail. These results are then used to characterize the fuel blistering, failure, and relocation processes indicated in Table 1. The available data provide a clear basis for characterizing the behavior during a severe accident.

BEHAVIORS OBSERVED DURING Al–U FUEL-MELTING TESTS AND INCIDENTS

The experimental observations presented in this section provide important information concerning the failure behavior of irradiated metallic Al–U alloys during a melting accident. Some of the data from these observations show the importance of the oxide film present on the surface of the molten fuel in determining the relocation mechanics of the molten alloy. The results of the observations presented in this article indicate that the fuel can fail in three distinct stages during a melting accident at power densities less than typical Al–U fuel power densities. This behavior is observed in several studies of fuel overheating that have been conducted as part of the safety evaluations of these reactors. Additionally, several incidents involving varying amounts of fuel damage have also occurred. A summary of these sources is given in Table 1. In addition to the data sources listed in Table 1,

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Table 1 Nuclear Metallic Aluminum–Uranium Fuel-Melt Behavior Data Base

Data source	Blistering	Cracking	Flow from cracks	Flow regime	Burnup
SRL out-of-pile tests	Not reported	Yes	Yes	Rivulet	No
SRL annealing tests	Yes	Not reported	Not reported	Not reported	High
ATR annealing tests	Yes	Not reported	Not reported	Not reported	High
SRS in-pile incidents	Yes	Yes	Yes	Rivulet–foam	Low
WTR incident	Not reported	Yes	Yes	Not reported	Low
SPERT tests	Not reported	Yes	Yes	Rivulet	None
SRL SPERT tests	Not reported	Yes	Yes	Rivulet	None

a large number of experiments were performed in the TREAT reactor. These experiments were conducted with small fuel samples and were used to study oxidation effects only.

The following sections of the paper describe the information available in more detail. The items addressed are blistering, fuel-cladding cracking–failure, and fuel relocation mechanics. The information presented provides a clear basis for the identification of the phenomena reported in Table 1.

Fuel Blistering Behavior

Annealing (a time at temperature process) of Al–U fuel plates and tubes has shown that fuel failure caused by blistering depends on temperature and, to a much lesser extent, burnup. Fuel failure occurs by excessive blistering during annealing 100 to 200 °C below the melting temperature. Fuel blistering is the result of the internal aluminum matrix cracking of the fuel. The internal gas pressure of the matrix causes the cracks to grow and thus allows the gas in the fuel matrix to exert pressure on the fuel-cladding interface. Blistering results when the gas pressure and clad temperature allow the clad to locally debond from the fuel. These observations have been established by posttest metallographic examination of annealed fuel specimens.

Several sources of gases in the fuel matrix are thought to influence blistering. These are gases absorbed during fabrication (dissolved hydrogen), gas-forming impurities in the matrix aluminum, and fission products. The onset of blistering is not a strong function of burnup. Thus it appears that gas forming impurities and gas absorption during manufacturing are major contributors to blistering. Several types of blisters have been observed. The types of blisters are grouped by size into three categories:

- *Small:* Separation occurs at the fuel-cladding interface. Blisters are typically 2 to 3 mm in diameter.
- *Medium:* Several millimeters in diameter. Similar to the small blisters formed at the fuel-cladding interface.
- *Large:* These blisters occur in the fuel core itself. Typical sizes are several centimeters in diameter.

Several annealing studies have been conducted at the Savannah River Laboratory (SRL) and at the Idaho National Engineering Laboratory (INEL)[1] on the blistering–swelling behavior of irradiated uranium and Al–U alloys. These studies with a high-burnup (~50%) aluminum–25 wt % uranium alloy indicated that annealing at 400 °C caused very little swelling. Temperatures from 475 to 550 °C resulted in extensive blistering and cracking of the cladding of the fuel. The in-pile overheating and melting accidents that have occurred at the Savannah River Site (SRS) have all shown evidence of blistering on the affected fuel assemblies.

The SRL annealing experiments indicated that the swelling of the fuel is low until about 10 °C below the melting point of the alloy. Above this temperature, extensive deformation is noted, and the release of fission gas begins to occur. Swelling occurs as a result of fission-gas bubble agglomeration on UAl_4 grains.

Two gas disruption failure modes are observed from the SRS experiments: coherent cracking and blistering. Large amounts of gas are expelled from the fuel as a result of the cracking and blistering. The releases take on the appearance of gas jets. The cracking propagates in the vicinity of the fuel-clad interface in a transverse direction to the fuel meat thickness. Blockages and perhaps inclusions are capable of deflecting the direction of crack propagation into the fuel meat. There is no evidence that the cracking is associated with local debonding of the clad from the fuel. Rather, the crack failure mechanism appears to be a consequence of classic stress failure across two different materials. The stress arises from forces in the fuel meat. These forces can be the result of internal gas pressure, material phase changes, and residual stress from the manufacturing process. Unlike cracking, the blister formation occurs randomly throughout the fuel meat.

Insufficient data exist from the SRS fuel-melting tests to definitively delineate cracking and blistering volume expansion. However, it can be stated that as a minimum the gas disruption failure results in a radial expansion of about 30%.

Clad Failure and Melt Flow Regimes

As indicated, metallic Al–U fuels can fail in three distinct stages during a severe core damage accident. These stages are associated with fuel that is at or less than nominal power. The first stage of fuel failure during a severe core damage accident is blistering. The second and third stages of fuel failure are clad cracking and relocation of molten fuel material through the cracked but still solid cladding. This behavior is a result of the difference in melting temperature between the fuel and cladding alloy. In general, the cladding used with Al–U fuels melts about 10 to 20 °C higher than the fuel. This melting temperature difference increases as burnup progresses. The increase in the difference between the melting points of the cladding and fuel is a result of the buildup of fission products (e.g., silicon). The addition of fission products to the fuel alloy tends to lower the eutectic point of the fuel.

An extensive experimental data base supports the cracking and fuel draining phenomena. Experimental information is available that provides these observations from a wide range of in-pile melting incidents. These data include the SRS overheating incidents and the Westinghouse Test Reactor (WTR) incident. In-pile fuel-melting experiments that have been conducted in the SPERT reactor using both plate and tubular fuel elements have failed in this manner. Out-of-pile experimental programs using tubular elements heated by induction have also failed in this manner. A description of some of the available information is given in support of these observations.

Simulated Fuel Tube Melting Studies. Several experiments were performed at SRS in which a 1-in.-diameter unirradiated Al–U tube clad with 8001 aluminum alloy was heated inductively to melting in flowing steam. These experiments indicate that the fuel's cladding fails by cracking, which allows the molten fuel core to flow out the cracks in the clad. The molten fuel flowed as rivulets down the surface of the fuel and thus froze in place as an agglomerated sheet on the unheated portion of the tube. Posttest examination of the tube indicated that cracking had occurred along the grain boundaries of the cladding. Clad cracking is postulated to have occurred as the result of grain boundary melting and pressure exerted by the 5 to 6% volume change that occurs upon melting of the fuel core. Frozen fuel droplets were found on the surface of the cladding. These fuel droplet observations indicate that the molten fuel was not wetting the surface of the clad.

Experiments were also conducted with unirradiated 1100 aluminum alloy tubes at SRS. These tubes were heated in an induction furnace to failure. Failure occurred before bulk melting. Tube failure resulted in the formation of axial through-wall cracks along the surface of the tube. Surface examinations of the tube indicate that the failure was due to grain boundary melting of the alloy. Large pieces of the tube dropped off and thus left openings to the inner parts of the tube.

In-Pile Postincident Fuel-Melting Analysis. At SRS californium-252 was produced by using special metallic Al–U fuel-element assemblies. These fuel elements consist of three concentric fuel tubes, each consisting of 6 ft of enriched Al–U alloy that are clad with 1100 alloy on the inside and 8001 alloy on the outside. The cladding and fuel are metallurgically coextruded together to form an adequate heat transfer bond to the fuel. These fuel elements were operated at a very high power to obtain a high enough neutron flux for production of californium. Typical assembly power levels were 20 to 25 MW. This power level was very close to the calculated burnout heat flux for the assemblies. Six of these californium fuel elements experienced localized melting during early 1970. Postfailure experimental analysis revealed that the fuel failures resulted from the formation of blisters, cladding holes, and cracked cladding with localized Al–U melting. Several other tubes showed signs of blistering and pitting. Rib marks were noted on the surface of the tubes, and melting occurred in patches parallel to the rib lines. Failure was due to burnout in patches on the cladding surface caused by rib effects. Heat transfer to adjacent tubes and coolant occurred through the ribs. The rib heat transfer effects limited melting to the center of some of the rib circles. Some of the failures resulted in fuel draining and freezing on the surface of the outer clad. Other fuel failures resulted in the molten fuel remaining in place and not moving from the failure location.

A metallic fuel assembly used at SRS to produce plutonium failed as a result of partial melting after 33 days of irradiation in early 1970. The failure involved more than 50% of the assembly. Fuel failure was accompanied by clad blistering, clad cracking, and the Al–U core melting and draining through cracks in the cladding. Postfailure analysis indicated that the fuel drained as rivulets on the surface of the cladding and froze at cooler locations that were not experiencing dryout. Fission-gas bubble agglomeration resulted in the expansion of the cladding material, which doubled the transverse thickness in some locations of the assembly. Agglomerated fission-gas bubbles were present throughout the molten fuel core material. These large gas bubbles indicate the onset of fuel foaming and significant swelling.

A metallic fuel element failed in the WTR (Ref. 2) in April 1960. This failed fuel element released fission products into the coolant system. Coolant boiling experiments were being conducted on this metal-fueled, water-cooled reactor before this failure occurred. These boiling experiments resulted in the melting of one high-powered fuel assembly in the core of the reactor. Postaccident examinations of this fuel-melting accident indicated that the fuel core melted inside the solid cladding and the molten core drained through cracks that had formed in the cladding. The molten fuel refroze in the coolant channel at a low power region of the assembly. The refrozen fuel formed a porous blockage in the bottom of the fuel assembly.

SRS's SPERT Test. Six unirradiated fuel tubes similar in design to early Savannah River fuel were melted in the SPERT I reactor in 1958.[3] These fuel tubes were 2 ft in length. The fuel, an aluminum alloy with 31 wt % uranium, was clad with 1100 aluminum alloy and was formed by coextrusion of the fuel and cladding. The coolant flow for the fuel tube was downward as in the SRS reactors. The experiments were conducted by placing the SPERT I reactor on a short period. This overheated and partially melted the experimental metallic fuel tubes. Visual and radiographic examination of the melted fuel tubes showed that the fuel material had melted and flowed out cracks in the cladding. This occurred because the melting point of the fuel is 20 °C lower than the melting point of the cladding used in these experiments. In none of these experiments did the spacer ribs of the fuel assembly melt. Three factors prevented the ribs and the local region near them from melting.

- In this tubular fuel design, the cladding is thicker at the ribs compared with the rest of the cladding.
- The ribs acted as an effective heat transfer medium, conducting and convecting heat to the surrounding tube and to the flowing coolant.
- The molten fuel flowed from the core beneath the ribs and thus reduced the heat source as soon as a crack in the cladding occurred.

The molten fuel drained as rivulets to the bottom of the test assembly in these experiments. Near the bottom of the assembly, the rivulets collected on a cold surface and formed an agglomerated rivulet sheet. This molten sheet then flowed as long drops, or jets, to the bottom of the test assembly. Some of this melt was entrained by the flowing coolant. For the most severe melting case in which about 25% of the fuel melted, about 7% of the molten fuel was recovered as a particulate. This particulate was in the form of jagged flakes and agglomerations of once molten particles.

SPERT Experiments. A series of experiments was performed in the SPERT I reactor to investigate the effects of rapid transients on the behavior of unirradiated metallic fuel plates.[4] The experiments progressed in energy yield to the point where a reactor core was substantially melted and destroyed by a steam explosion. Several conclusions were drawn from the SPERT experimental data by the investigators. The most important conclusion relative to metallic fuel behavior was that, subsequent to thermal distortion, molten fuel escaped through cracks in the unmelted clad. Molten fuel material in these series of experiments flowed as rivulets down the surface of the fuel plates, and the rivulets were thick enough to bridge the coolant gap between adjacent fuel plates in many cases. The gap thickness in these experiments was of the order of 5 to 6 mm.

TREAT Experiments. The TREAT facility is an experimental test reactor located at INEL. This facility was used to assess the behavior of small samples of unirradiated metallic fuel plate subjected to rapid heating transients.[5–10] Several experiments were conducted in this reactor with small metallic fuel plates. These fuel plates were Al–U alloy clad by an alloy of aluminum. The reactor periods that drove the fuel to destruction ranged from 0.108 to 0.285 second.

The small metallic fuel plates that were subjected to moderate energy inputs fused into a sphere. Peak temperatures in these experiments were less than 1200 °C. The fuel plates in these experiments became incandescent and chemically reacted with the water surrounding the molten fuel. At the highest energy inputs the fuel samples ignited, and sustained burning of the sample under water occurred. Molten fuel temperatures in these tests exceeded 2000 °C. Extensive fragmentation of the molten fuel also occurred during these high-energy transients.

In almost all these experiments the small sample of fuel and cladding material fused into a uniform spheroidal molten mass. Strong evidence of the effects of surfac tension was seen upon melting. In the very rapid heating transients the fuel plates collapsed into a thin molten cylinder as the fuel melted and then began to fall. In some of these experiments the interaction of the fuel with the coolant produced extensive voiding of the fuel such that the fuel floated on the surface of the water.

Measurement of the Fluid Point of the Fuel Alloys

One experiment was conducted at SRS to determine the fluid point of molten fuel alloy. In these experiments droplets of unirradiated 35 wt % Al–U alloy were cast and suspended from a balance. These droplets were then heated to melting, and the fluid point (relocation) temperature was measured. These experiments demonstrated that the alloy begins to flow near its eutectic point. As a result, the fuel alloy behaves in the same manner as a homogeneous solid upon melting (that is, near the eutectic point of the alloy, the temperature of the alloy remains constant and the molten Al–U alloy begins to drain with little superheat). If the temperature increases until the phase transitions of the Al–U compounds begin, the temperature of the liquid remains constant until sufficient energy is absorbed to complete the transformation of the Al–U intermetallic compounds to the next metallic phase.

An analysis of the SRS experiments by Ellison and Monson indicates the following conclusions:

- A superheat of 10 °C is required for gross movement of the molten alloy. This observation is in agreement with the behavior of most other aluminum alloys in which fluidity is found to be a function of the solidus–liquidus partition function.
- Because these tests were conducted with unirradiated alloy, no statements regarding the behavior of irradiated alloy can be fully supported by these tests. (Note: The SRS irradiated fuel-melting experiments appear to indicate that the fluidity temperature of foamed fuel is greater than that of unirradiated fuel.)
- Embedded thermocouples in the irradiated SRS melting tests have shown that the eutectic temperature of the irradiated alloy is about 642 °C. This eutectic is only slightly smaller than the unirradiated eutectic of 646 °C.

Experimental Assessment of Irradiated Fuel-Melting Behavior

The dimensional stability of the fuel is important to reactor safety during irradiation and during postulated accidents. Fuel swelling can lead to coolant channel thickness reductions and fuel overheating. Aluminum–uranium based fuels are highly stable and generally do not noticeably swell during irradiation at nominal temperatures. During a core damage accident it is important to assess the geometric stability of the fuel. This assessment is needed to determine the ability of the Emergency Core Cooling System (ECCS) to reflood an overheated core. Fuel swelling during heatup as a result of insufficient coolant can lead to conditions where it is not possible for ECCS water to reenter and cool a reactor because of coolant channel thickness reductions. Several phenomena are important to understanding the ability to reflood a badly damaged but unmelted core. Some of the nonhydrodynamic processes are related to the fuel swelling phenomena. These are fuel swelling processes and important parameters:

- Blistering
  - Temperature of onset
  - Rate
- Fuel Foaming–Swelling
  - Temperature of onset
  - Rate

Several annealing experiments were conducted on highly irradiated Savannah River fuel to understand the rate effects and temperature dependencies of the geometric stability of the fuel during a heatup–melting transient. Several types of experiments were performed. These tests annealed high-burnup metallic and cermet Al–U alloys. Several other experiments were conducted to determine the strength of the oxide film present on the molten layer. The oxide film serves to increase the surface tension of the molten alloy. This enhanced surface tension results in an increase in the molten mass required to allow rivulets to move or break away from the melt-out region.

Metallic Al–U Experiments

These experiments were scoping in nature to determine the general nature of undercooled, irradiated, metallic Al–U fuel behavior. The fuel used in these experiments was a 33 wt % alloy of uranium with aluminum and clad with 8001 aluminum alloy. The burnups of the fuel samples used in the experiments varied from 45 to 50%. The fuel sample masses were of the order of 10 g.

The experiments were conducted in the Savannah River high-level caves in middle to late 1991 and early 1992. The testing program involved the annealing at different rates of irradiated Al–U coupons about 1 in. square. These coupons were cut from fuel assemblies irradiated in an SRS reactor in 1982.

Three different types of experiments were conducted in a mockup of a typical SRS fuel assembly geometry. In the first series of experiments, the geometry was chosen so that an irradiated fuel coupon was annealed in between two aluminum plates. The channel thickness between the fuel coupon and the aluminum plate was maintained to be typical of an SRS coolant channel thickness. In another series of experiments, three fuel coupons were used with a similar gap thickness. The third set of tests involved supporting a fuel coupon from either the top or the bottom to determine flow characteristics of the melts.

These coupons were supported by a graphite holder and heated to melting in a flowing argon atmosphere inside a quartz tube. Heat was supplied by convection and radiation through a tube furnace. A video camera was placed inside the hot cell so that the channel gap could be observed during the annealing. The video record was also used to observe the swelling and blistering behavior of the fuel. Measurements were also made of the beta activity of the flowing argon gas and the temperature of the furnace atmosphere. However, no fission-product element release rate measurements were taken. Typical temperature heatup rates were varied from 0.2 to 1 °C/s.

Several key observations were made from these experiments.

- The oxide film present on the surface of the melt strongly enhances the surface tension of the molten alloy and influences the flow behavior.
- Little swelling is observed until the fuel specimen is near its eutectic point; then rapid swelling is observed.
- The swollen–foamed condition is transient. The stability of the foam is measured in terms of seconds to minutes. Upon foam collapse the alloy seems to drain into a porous molten pool inside an oxide shell. Inside this shell the phases appear to separate partially. The gas periodically vents through self-healing cracks that form in the oxide shell. The oxide shell then completely collapses into the collapsing foam.
- Blistering was observed to occur rapidly above 550 °C when it occurred. Blistering did not occur on all the annealed coupons.
- Relocation of the alloy as a film or rivulet did not occur. The mass of these fuel coupons was insufficient to overcome the oxide-film-enhanced surface tension to allow the molten alloy to flow.

These experiments also indicate that the swelling rate is a function of the rate of temperature change. The total amount of swelling was found to be consistent for all the experimental geometries. For both horizontal and vertical geometries, the average estimated foam porosity was around 63%.

Several in-pile irradiated fuel-melting experiments have also been funded by the U.S. Department of Energy. These experiments were conducted by the Sandia National Laboratories (SNL) for the Heavy Water New Production Reactor Program. These SNL experiments melted in-pile long sections of an irradiated SRS Mark 22 tritium production assembly. The preliminary results of this experimental program noted that significant foaming of the irradiated Al–U fuel alloy was occurring.[11] The results of these experiments are in general agreement with the SRS hot cell experiments and previous accidents in Al–U fuel reactors.

For accident sequence analysis, it is important to note that extensive swelling of the fuel was not observed until just before the fuel melted. This delay in swelling and the resulting subsequent delay in the coolant channel flow reduction until the onset of melting indicates that ECCS flow degradation should not be influenced by foaming and swelling until fuel melting begins. In contrast, blistering needs to be addressed because it occurs rapidly and can lead to coolant channel flow restrictions before fuel melting occurs. However, blistering is difficult to predict because it is a function of the manufacturing process, the quality control procedures used in the production of the fuel, and the temperature and burnup history.

Melt Behavior of Cermet Al–U Fuels. Several annealing experiments using irradiated cermet fuels were conducted at the Westinghouse Savannah River Technology Center Shielded Cells. Cermet fuels are of interest in the production of isotopes because they allow for higher uranium densities than alloy fuels. The cermet Al–U fuels consist of a dispersion of U_3O_8 in an aluminum matrix clad by an aluminum alloy. The alloy used in the Savannah River tests consisted of 28.7% uranium before irradiation.

Two characteristic annealing studies were performed in which the melting behavior of the unirradiated cermet fuel and irradiated Al–U alloy fuel were compared with melting behavior of irradiated cermet fuel. The burnup of the irradiated fuel was more than 50%. The experimental procedure was identical to the earlier SRS Al–U alloy fuel coupon melting studies. The following observations summarize these melt studies:

- The irradiated cermet fuels do not foam.
- The irradiated cermet fuel exhibits extensive blistering at the fuel-cladding interface (i.e., debonding of the cladding). Typical blister thickness ranges from 50 to 100% of the total initial fuel coupon thickness. The blistering involves the entire cladding and is representative of a nonlocalized phenomena.
- The unirradiated fuel shows little evidence of the gross blistering observed for the irradiated cermet fuel. The blisters are localized with thickness less than 25% of

the total initial coupon thickness. Surface eruptions are apparent from the molten fuel before mechanical failure of the fuel coupons.

- Upon the failure of the irradiated cermet fuel blisters, a peak is detected in the radioactive fission-gas release from the fuel.
- The melt failure of the unirradiated cermet fuel is defined in terms of a melt slumping of the aluminum matrix.
- The irradiated cermet fuel failure sharply contrasts with that of the unirradiated cermet fuel in that only the clad is observed to slump away from the fuel meat. The irradiated cermet fuel meat remains a rigid body up to about 1200 °C.
- The irradiated cermet clad begins to slump off the fuel meat approximately 20 K below the temperature when the unirradiated cermet clad loses its structural integrity. The result is melt failure of the entire coupon.
- The temperature of the clad blister collapse and subsequent clad slumping of the irradiated cermet fuel and the mechanical failure of the irradiated Al–U fuel clad coincide. Foaming of the irradiated Al–U metallic alloy also occurs on a time frame coincident with the clad blister initiation of the irradiated cermet fuels.

Comparisons of the unirradiated and irradiated cermet fuel melt studies indicate that much of the aluminum matrix must be converted into a metal oxide as a result of irradiation to explain the observed difference between melt slumping and structural rigidity, respectively. Also, as a result of irradiation, oxide transformation, and subsequent annealing of the fuel matrix, extensive fission-gas transport to the fuel-cladding interface occurs, which results in large blisters. Conversion of the aluminum matrix into an oxide elevates the melt failure temperature of the cermet fuel to a value characteristic of an oxide.

Influence of the Pliable Oxide Film. Several experiments were conducted at the SRL to understand the importance of the oxide layer of the molten fuel on melt relocation mechanics. As noted during previous experiments, the oxide film was controlling melt relocation behavior. Several tests were performed with molten irradiated metallic Al–U alloys in which various weights were placed on the surface of the oxide film to increase the stress on the surface of the oxide. Under sufficient loading the molten fuel was observed to slowly move outward from the cut ends. The melt progression corresponded most closely with the formation of a large high-surface-tension droplet. In experiments with light loads, little melt relocation has been observed before mechanical melt failure of the clad.

Upon melt failure of the fuel coupons, relocation of the molten fuel was restrained by the presence of the oxide scale that formed an encasement or shell about the melt. The gross movement of the molten fuel did not appear to break the oxide shell, rather two species, the molten fuel and the oxide shell, appeared to move in concert together. It is probable that, at these melt temperatures, self-regeneration of the oxide scale is sufficiently rapid to self-heal microcracks and fissures that result from the gross movement of the oxide shell. As a result, the otherwise brittle oxide shell takes on a plastic appearance. Relocation of the molten fuel into the graphite crucibles did not result in the melt filling out the available cavity space; rather the oxide-encased molten fuel tended to ball up in accord with a high-surface-tension material. Further heating of the molten fuel resulted in the periodic venting of hot gases via rapidly self-healing fissures within the oxide shell.

Rivulet Flow Regime

The previous review of the in- and out-of-pile melting experiments and incidents noted that the molten alloy tends to bead up on the surface of the cladding as droplets. In those incidents in which a small localized burnout occurred or a small region overheated as the result of clad debonding, the molten fuel material did not move from the site of the melting. In other cases the molten material moved as rivulets near the ribs of the assemblies or bridged large gaps between adjacent fuel plates. This behavior indicates that the fuel alloy does not wet the surface of the cladding. Molten fuel may not spread on the surface of aluminum-clad fuel elements because of the nonwetting of molten aluminum on aluminum oxide.[12] In these nonwetting conditions, the fuel melt will drain, and the leading and trailing edge of the fuel deposit will make an angle with the solid surface.

Rivulet flow is the expected flow regime for molten Al–U. Its motion is limited by both frictional resistance and interfacial-surface-tension forces. Molten fuel rivulet drops will not drain until their mass exceeds a critical value. As a result, it is possible under conditions of localized burnout that the molten material will remain in place and not relocate. This type of behavior was seen in the SRS californium charges in which burnout occurred. The thickness of the fuel-melt front can be sufficient to bridge the gap between fuel plates or fuel tubes. This gap-bridging behavior occurred in a melting accident involving a plutonium production fuel driver at SRS. The molten fuel material needed sufficient mass that the coolant gap between the fuel tubes was bridged before the molten fuel could drain. As a result of this bridging, the molten fuel material was cooled by coolant flow on the outside of the intact fuel tube. The molten mass then froze in place, and the accident was terminated.

This critical mass value for relocation of a molten rivulet plays a very important role in in-core accident progression and analysis of Al–U fueled reactors. This critical relocation value of the molten mass is a strong function of the receding and advancing contact angles along with the surface tension. The surface tension may be enhanced by a thin, pliable, oxide film that forms on the surface of the Al–U alloy. The higher the surface tension, the larger the amount of mass required before the rivulet can move. Molten rivulets below the critical relocation mass may remain in place, without moving, until they are captured by an advancing melt front. The rivulets will move downward until their mass falls below the critical value and will then stop and freeze in place. The reduction in mass of the rivulet as it moves is caused by the freezing of the molten material of the rivulet on the surface of the cladding.

Foam Flow

Metallic foams can occur in fuel melts.[13] Foams form from the agglomeration of fission gas and other gas bubbles in the melts. Experimental data from out-of-pile testing with irradiated uranium and Al–U alloys indicate that foams do occur. Foams of metallic uranium were noted to form in the following manner:

- Cracks form in the fuel specimen and rapidly increase in length and width.
- The cracks appear to grow in width more than in length.
- When the metal begins to melt, the jagged edges of the cracks become smooth.
- Bubbles form and then become round because of surface-tension forces.
- Bubbles then grow by agglomeration forming a low-density foam.

In the SRS fuel-melting experiments, foam expansion results in gross volumetric expansion and the filling of the voids of the cracks. The cracks first rapidly propagate lengthwise. The gas release coupled with thermal stress relief causes the cracks to widen. The dimensional expansion is of the order of 25 to 30%.

The SRS in-pile melting evidence indicates that metallic foams occur with burnups greater than 1 at. %. A low-burnup (less than 2 at. %) metallic fuel element that failed because of localized overheating at SRS experienced the onset of foaming. Large agglomerated fission-gas bubbles were seen in the postincident metallographic examinations. A foam had begun to form in the fuel and then expanded, which forced a metallic fuel foam through the cracks of the cladding. The molten metallic foam appeared to flow as a rivulet.

Molten Fuel Entrainment

Molten metal on the surface of a solid can be entrained into the coolant if the coolant velocity is sufficiently high.[14–17] Above a critical gas velocity molten fuel will be entrained as droplets in the flow. For these high-surface-tension metallic fuel metals, the entrainment mechanism is not well understood. The material dynamic strength properties of binary alloys, such as grain boundary fragmentation of the fuel, were thought to be responsible for the entrainment in some of the previously referenced SRS experiments. Some of the SRS experiments used very high velocity gas flows (sonic). These high gas velocity flows are not typical of that expected during a fuel-melting accident because the gas or liquid velocity is pressure-drop limited outside the fueled region of the fuel assembly. The particle size measurements of the SRS tests are not in general agreement with the results of other data and the models of interfacial stability relating to particle breakup. The reasons for the lack of agreement between the experiments have not been determined. However, the difference may be related to the grain boundary dynamic strength properties of the Al–U fuel alloy near its solidus point and the high gas velocity used in the SRS experiments.

FISSION-PRODUCT RELEASE

The melting behavior of the fuel is a strong function of the fission-product release during the heatup to melting. The temperature transient is also extremely important. Rapid heatup transients tend to contain a large fraction of the fission-product inventory of the fuel within the fuel before melting. These high ramp rates lead to significant fission-product-induced swelling and foaming once the alloy fuel becomes molten.[18] In contrast, the fuel does not experience extensive foaming or extensive geometric expansion upon melting in those transients in which the majority of the volatile fission products are released before melting.

The melt behavior attributed to foaming is a function of the amount of volatile fission products in the fuel at the onset of melting. The parameters that influence the release are the type of fuel, the burnup, and the heating rate. As indicated, the timing of fission-product release influences both the fuel relocation mechanics and the subsequent fuel pool behavior. The experimental fission-product release data base that can be used to understand the relocation behavior of the fuel is extensive. Sufficient data exist to provide for a qualitative understanding of the behavior of the fuel. The fission-product release information supports and complements the molten fuel behavior just described. The fission-product release experimental data are, however, lacking in respect to detailed information that can be used to accurately predict the radionuclide source terms under all conditions of burnup and atmospheric compositions.

The items of interest in understanding fission-product and fuel behavior are the fission-product chemistry, fission-product release from the solid state, and fission-product release from the molten state. These three items are discussed in the following sections.

Fission-Product Chemistry Data

The rapid release of fission products from the fuel at the onset of melting drives the fuel foaming process. The volatile fission products contribute to the rapid expansion of the fuel upon melting at the eutectic point. Fission products to be considered in this process are xenon, krypton, iodine, and cesium. The reactions of these fission products with each other and the alloying elements in the fuels to produce volatile compounds all contribute to the foaming process.

The metallic Al–U fuels consist of a UAl_x particulate in a matrix of aluminum alloy. During irradiation fission products are released from the particulate but do not diffuse far from the UAl_x particulate. During normal irradiation and during accident conditions, the fission products are released from the Al–U particulate into a solid or liquid matrix of aluminum over a wide temperature range. The interaction of the fission products with the aluminum matrix is important in assessing the transport of the fission products through the alloy to the surrounding gas phase.

The fission-product chemistry theories that are needed to assess the phenomena that govern the fission-product release rates exist. However, few supporting data are available to use the chemistry models. There are some data on the reaction of some of the fission products with aluminum. Few of the data required to perform numerical evaluations of the vapor pressure of the species above the melt and their solubilities in the melt are available.[19,20]

The computation of the fission-product releases and the rate of release during a severe accident requires basic chemistry data in the following areas:

- Diffusion coefficients (solid, liquid, and gas).
- Solubilities of fission products.
- Phase diagrams of the fission products.

Diffusion coefficients exist for the alloying species that are used in aluminum-based alloys.[21–23] Almost all the diffusion data are found in the solid phase. Experimental work has been performed to measure the diffusion coefficients of elements in liquid metals.[24] Analytical expressions have been formulated on the basis of molecular theories. These theories do quite well in predicting the diffusion coefficients of elements in liquid metals.[25–27] There is substantial information on the diffusion of vapors in gases.[28] However, there are few data on the diffusion of fission-product gases in air or steam.[29]

An important aspect of understanding the release rates of the fission products is the solubility of these elements or compounds in the aluminum and/or Al–U particulate. A large amount of data exists for the elements used to alloy the aluminum,[30] and a few data points are available for fission products in aluminum.[31] Almost all these data are restricted to a very limited temperature range.

The phase diagrams of the most important fission products in aluminum are available.[32] However, there are no reported data on the ternary phase diagrams of the fission products with the Al–U alloys.

Iodine is an important contributor to fuel performance during a severe accident because it is one of the more volatile elements released from the fuel. Experimental data suggest that iodine is released at the same rate as the noble gases. As a result, it can contribute to the fuel foaming–swelling process. The exact chemical compound distribution of iodine in the metallic melts is yet to be determined. During the rapid expansion of the fuel, any elemental iodine in the gas phase would see a large surface area of molten alloy. This, in turn, allows the iodine to interact with the alloying elements and other fission products. This uncertainty over the exact chemical form of iodine is different from the light-water-reactor (LWR) case, in which it has been determined that cesium iodide (CsI) is the dominant chemical form of iodine. Iodine in metallic fuels has been found to form compounds with zinc, uranium, aluminum, and cesium.[33,34] Experimental evidence has also suggested that some iodine released from the fuel is easily transportable in the gas phase.[35] This ease of transport suggests that it is being transported as molecular iodine or as submicron aerosol particles. This conclusion is the result of iodine being deposited in the off-gas filters of some experiments rather than in the thermal gradient tubes of the experimental facilities. The formation of metallic iodine aerosols (ZnI) in these experiments could supply a sufficient reason for the iodine to be found in the filter media of the experimental facilities.

There is uncertainty over the exact chemical compound distribution of iodine released from solid or molten fuel. Evidence points to the formation of metallic iodine compounds with some data indicating the potential for molecular iodine. The molecular iodine theory can also be replaced by an equivalent theory describing the formation of a metallic aerosol compound in the experiments in which molecular iodine was thought to exist. The most likely form of iodine is a metallic iodide compound with cesium, aluminum, uranium, zinc, or another trace metallic alloying element. The potential for forming molecular iodine exists, but sufficient data do not exist to completely characterize the amount of it compared with the other metallic compounds. Iodine is thought to exist in the Al–U melts as CsI, UI_3, AlI, or ZnI.

Elemental cesium is another fission product of interest that may contribute to fuel foaming. Elemental cesium is insoluble in aluminum and uranium and also is volatile at the melting temperature of the alloy. It exists as a separate phase in aluminum and uranium melts. It does not form metallic compounds with either of these two elements. As a result, elemental cesium is readily vaporized from the molten fuel because of its high vapor pressure and its insolubility in the molten fuel alloy. During this fuel foaming stage the elemental cesium vapor may react with elemental iodine in the agglomerated gas bubbles in the molten alloy. Cesium can also form low vapor pressure compounds with silicon in the melts.[32] The formation of these compounds can reduce the volatilization rate of cesium compounds from the fuel alloy melts. There are no data on the Gibbs free energy of the cesium silicate compounds in the aluminum or uranium melts.

Also of interest to understanding fuel performance are the strontium and barium compounds. These elements have sufficient high temperature volatility in their elemental state to be of interest. Both strontium and barium, however, are known to form metallic compounds with aluminum.[36–40] The solubility of these two compounds in molten aluminum is relatively high.[30] Thus the potential to vaporize barium and strontium metals from the molten fuel alloy is limited after reaction with the aluminum matrix of the fuel.

Fission-Product Release From the Solid State

The foaming potential of the fuel is a function of the volatile fission-product inventory when the fuel melts. The ability of the fuel to foam or swell then is a function of the radionuclide release rate up to the point the fuel becomes molten. The information that is available on the release of fission products from the solid state is extensive. The data base consists of the work of Oak Ridge National Laboratory (ORNL), Reynolds of Knolls Atomic, the data of Dienst, the work of Shibata, and the data of Woodley at Hanford Engineering Development Laboratory (HEDL).

The experimental data are sufficient to make quantitative predictions. These predictions indicate little fission-product release during the heatup process. This lack of inventory release allows fuel foaming to occur. These predictions make use of typical heatup rates and empirical correlations of the fission-product release on the basis of the data described previously.[18] The following sections provide the basis for this observation. The information from the observations of the fission-product release is in agreement with the information from observations of the behavior of the fuel once molten.

Reynolds' Data. The release of krypton from 20% burnup irradiated Al–U alloy was investigated by Reynolds in 1958.[41] His experiments were conducted at temperatures between 878 and 968 K. In these series of experiments, little release of krypton was measured for temperatures below the eutectic point of the alloy. The conclusion was reached that the aluminum matrix surrounding the Al–U particulate is not permeable to gas diffusion until the matrix of the fuel melts. At temperatures above the eutectic point, the evolution of krypton from the fuel samples followed the expected time dependence of gas diffusion from spherical particles. Reynolds also attempted in these series of experiments to thermal cycle a fuel sample to enhance the gas evolution rate. This attempt to cause cracking in the aluminum matrix by thermal cycling was not successful.

Reynolds concluded that the Al–U particulate releases the krypton to the aluminum matrix at all temperatures. However, the aluminum matrix is impermeable to krypton diffusion until the matrix becomes molten. This result is inferred by Reynolds from the agreement of his data from two different temperature histories.

The results of Reynolds indicate that the release rate of the fission products is small until the fuel matrix melts. The fission-product release is being limited by the impermeable aluminum matrix that contains the dispersed UAl_x particulate.

Shibata's Data. The fission-product release rate from irradiated Al–U fuels was also measured by Shibata and coworkers.[42] These experiments were conducted with high-burnup metallic fuel plates clad with 6061 aluminum. The results of these experiments are in general agreement with the work of Reynolds. The results of Shibata indicate that the release rates of the fission products are small until the fuel matrix melts.

Shibata was able to categorize the release rate from these experiments in three stages associated with (1) blisters forming on the fuel, (2) melting of the 6061 cladding, and (3) melting of the fuel matrix. The data consist of release rates for xenon, iodine, and cesium as a function of temperature. The release rates appear to agree with the work of Reynolds with the additional releases occurring because of the formation of blisters and melting of the cladding. The fuel's 6061 aluminum alloy cladding has a lower melting temperature than the fuel alloy used in these experiments. Other fuels use an 8001 aluminum alloy cladding that melts at a higher temperature than the fuel alloy. The key result from the work of Shibata is that the fission-product release rates are small until the aluminum fuel matrix melts.

The data of Shibata also indicate the presence of clad blistering. Clad blistering was noted to occur in several of the in-pile incidents described previously and in some of the annealing studies conducted at the INEL and SRL. The data of Shibata indicate that the blistering process is associated with an increase in fission-product release. This increase indicates that the blistering process is, in part, associated with the behavior of fission products in the alloy.

ORNL Studies. At ORNL in the early 1960s an extensive fission-product experimental data base was developed from Al–U fuels.[33,43,44] The data include information as a function of temperature and atmospheric composition. The fission products for which data were obtained include xenon, krypton, iodine, cesium, and ruthenium. These data are in general agreement with the data of Reynolds and Shibata on the phenomena governing major increases in the rate of release of the fission products.

An important aspect of the early ORNL work was an experiment involving the effects of different burnups. Several tests were conducted on fuel coupons with different fuel burnups. After a certain burnup was obtained the release of gases from the molten fuel coupons became abrupt. This behavior was also noted in the SRL fuel-melting studies. In the case of the SRL tests, the abrupt releases were caused by rupture of the oxide film on the surface of the molten alloy. The abrupt release of fission products is an indication that the nucleation of vapor bubbles in the melt is occurring. These fission-product vapor bubbles provided a mechanism that allowed for the rapid evolution of the fission products from the molten fuel coupon. The nucleation and coalescence of vapor bubbles provides the mechanism for the fuel to swell and foam as described in the previous sections.

HEDL Studies. Fission-product release data have also been obtained for irradiated SRL Mark 16 fuel-element coupons.[35] These experiments were conducted at HEDL by Woodley. These experiments measured the release rates of noble gases, iodine, cesium, and tellurium in the atmospheres of steam, air, and argon. The temperature of the coupon was varied from 973 to 1373 K. These temperatures indicate that the fuel alloy was molten. Both cesium and tellurium were released at a faster rate in these experiments than in the earlier ORNL data. The data of these tests generally support the previous experimental observations of a solid impermeable aluminum matrix preventing fission-product release until the aluminum matrix melts.

Fission-Product Release Data Summary. The review of the fission-product release data from solid Al–U fuels provides guidance in understanding both the fuel swelling process and the release of fission products. The overriding effect for the noble gases, cesium, and iodine is the increased mass transport that occurs when the fuel melts. For temperatures below the eutectic point, the release fraction from the fuel is small. These small release fractions (less than a few percent) can be modeled by diffusion from the UAl_x particulate in the fuel. For high fuel burnups, consideration for blistering is required to properly model fission-product release. Careful attention also must be paid to the modeling of UAl_x particulate dissolution that occurs between the solidus and liquidus temperature of the Al–U alloys.

The fission-product release data indicate that the fuel foaming process occurs as a result of the holdup of the fission products near the UAl_x grains by an impermeable aluminum matrix. Once melting of the aluminum matrix occurs, the fission gas nucleates into small bubbles. These bubbles then agglomerate and thus cause the fuel to foam. The fission products are then released as the bubbles rupture a solid oxide film present on the surface of the alloy.

The available fission-product release data from the previous series of experiments have been compared.[45]

Wide scatter in the release rate data was noted between the different series of experiments. This scatter was attributed to differences in fuel burnup history. Observations of the fuel swelling experiments performed at SRL indicate that the fission-product release rate for molten coupons is significantly influenced by fuel swelling and bubble coalescence. The fission-product bubbles coalesce into large bubbles that burst through a thin but pliable oxide film on the surface of the alloy. The fission-product release was enhanced when this process occurred. The solid oxide film rupture process was random in nature. Also note that, because of the random nature of the oxide film rupture process, the measured fission-product release rates of the SRL experiments had significant time variations.

Fission-Product Release From the Molten State

The release of fission products from molten pools is an important aspect of fuel performance during a severe accident. The formation of liquid pools of fuel alloy is an expected condition for most hypothetical melting accidents. The alloy may consist of a two-phase solid–liquid slurry that becomes fully molten as the temperature of the melt increases to its liquidus.

The information required to understand the release of fission products from molten pools is gleaned from melt refining operations.[46-49] These melt refining operations are used in the production of the common metallic alloys. Experience with melt refining has shown the existence of the following phenomena that govern the removal of dissolved gases from the molten alloys. Dissolved gases are removed from molten metals through the following:

- Vapor evolution by diffusion of the gas to the surface of the melt.
- Vapor evolution by diffusion of the gas to bubbles in the melt.
- Vapor deposition in the slag or skull by diffusion to the crucible or skull region of the melt.

All these processes are expected to occur during a fuel-melting accident. Information on the behavior of fission products in molten pools is also provided by a new series of data on this subject. Several molten pool experiments were conducted using simulated Savannah River high burnup fuel. These experiments were designed to measure the release of fission products from molten pools of Al–U alloy. The following section provides an understanding of the macroscopic fission-product phenomena that influence fuel performance.

The evolution of fission-product vapors from the surface of the melt is strongly influenced by the formation and coalescence of bubbles in the melt. Release rates that occur by liquid-phase diffusion are very small compared with the release rates that occur if bubbles form in the pool. Bubble formation in the pools can occur spontaneously if the pool's superheat is sufficient to nucleate bubbles in the melt. Experience with degassing by the spontaneous formation of bubbles indicates that very high vapor pressures are required to form a bubble nucleus in a liquid metal. Pressures as high as 10 000 atm are required in some liquid metals.[48] The possibility of forming bubbles is enhanced if the melt contains suspended slag particles or has a large crucible surface area with sufficient sites to allow heterogeneous nucleation to occur.

The insoluble fission products in the melt nucleate into vapor bubbles near condensation sites. The condensation sites are typically considered to be the solid UAl_x particulate or other solid insoluble alloying elements or fission products. The fuel melts can contain suspended Al–U particulate when the temperature is between the solidus and liquidus points. The fission-product vapor nuclei coalesce into larger bubbles and allow the fuel to swell or foam. These considerations lead to the conclusion that the fuel melts will contain bubbles of nucleated fission products that are attached to the Al–U particulate in the melt. This phenomenon has been observed in the previously mentioned fuel-melting foaming studies.

The fuel melts are internally heated by fission-product decay. As a result, strong thermal convective currents can form in the melts.[47] The extent of the convective currents is a function of the Rayleigh number and the geometry of the melt. These convective currents have been observed in inductive melt refining operations. Melt refining theory is able to predict the influence of convective flows on the release rate of dissolved gases. An extensive experimental data base exists on the convective behavior of internally heated melts in different geometries and boundary conditions. An excellent review of this data base is provided by the doctoral work of Paik at the University of Wisconsin.[50] This data base can be applied to predict the heat transfer rates to the crucible and the convective flow of bubbles to the melt's skull or surface oxide layer.

Experimental data specific to fission-product release from metallic melts are available from the early work on reprocessing metallic uranium by melting and from the ORNL data. The reprocessing data are useful for accessing the expected fission-product release behavior in Al–U melts. The reprocessing work was reported in 1961 by Argonne National Laboratory (ANL). The release rates from uranium melts of

Ru, Mo, Pd, Rh, Tc, Sb, Cd, Te, Y, Ba, Sr, Cs, I, Kr, and Xe were measured in these experiments.

The results of these early ANL reprocessing–melt refining operations indicated that the noble gases, xenon, and krypton were 99% released upon melting of the fuel. The remainder of the noble gases remained in solution.[51]

Iodine acted as a metallic compound in these melt refining experiments.[34] Iodine was held up in the melt until the melt reached a temperature of 1673 K, after which it was released from the uranium alloy. No evidence for the evaporation of iodine as free iodine was found in these melt refining operations. The iodine in these experiments was possibly bound with uranium as UI_3 or with another fission-product element.

Barium and strontium in these experiments exhibited similar behavior.[52] There was little release of these elements from the melts. This indicates that substantial evaporative release of these two materials does not occur. A few percent of these elements were found in the slag around the inner walls of the crucible and in the skulls of the melts. This result indicated convection-enhanced mobility of the materials in the melt but no evaporation from the surface of the melt.

Cesium in these melt refining experiments was released by vaporization.[53] This was consistent with its insolubility in uranium and its high vapor pressure at the temperatures of the melt.

Tellurium in these experiments apparently formed a low vapor pressure compound and was not readily released from the melt.[54] As was expected, cadmium was readily vaporized from these melts. The remainder of the fission products, Ru, Mo, Pd, Rh, and Tc, were retained by the melt with no appreciable vaporization. These elements appear to be acting as elemental units in the melt and are vaporized according to their elemental vapor pressures.

The ANL conducted a series of experiments for SRL.[55] These experiments used Al–U alloy that contained the expected amount of fission-product elements for end-of-life SRL conditions. These experiments used Al–U alloy that contained the expected amount of fission-product elements for end-of-life SRL conditions. The samples were fabricated with depleted uranium and nonradioactive fission-product elements. The alloys were cast and fabricated into slugs that were remelted in an induction furnace. The amount of the fission-product elements released from the melts was then measured.

The SRL–ANL simulant experiments indicated that significant deposits of barium and strontium were found only at the two higher melt temperatures studied, 1189 and 1324 K. The activity coefficients for barium were calculated to be 5 and 15; for strontium, the activity coefficients were calculated to be 0.01 and 0.25 at the two indicated temperatures. These activity coefficients are considered to be approximate only.

For barium, the high activity coefficient indicates that the release rates will be greater than that estimated on the basis of ideal solution behavior. The opposite is true for strontium. Support for a low activity coefficient for strontium is provided by the results of a recent study of the aluminum–strontium system.[56] An activity coefficient of 0.024 was reported for a 0.17 mole fraction of strontium in aluminum and 0.005 for 0.091 strontium mole fraction. Note that strontium is strongly stabilized in solution by the formation of Al_2Sr, which could be the reason for the low activity coefficient of the strontium.

When cesium iodide was added to the mixture to be melted, the observed behavior of iodine indicated that the CsI was not soluble in the melt. The CsI release continued for over an hour at a rate that corresponded to an apparent vapor pressure much greater than that expected for a solution.

The behavior of the CsI in the gradient tubes was complex. The bulk of the deposit occurred in the temperature range from 730 to 900 K and appeared as fine crystalline deposits. Oak Ridge investigators reported collocations of CsI deposits in the 600 to 800 K range in experiments with irradiated LWR fuels.[57] In the SRL–ANL study, powdery white deposits containing important quantities of cesium and iodine were also found at locations corresponding to gradient tube temperatures less than 550 K. The deposits at 730 to 900 K and at 300 to 550 K were distinct and separated spatially. The ratios of cesium and iodine varied significantly along the gradient tube. Also, a trace of iodine, but no cesium, was found in the bubbler solution. Traces of iodine in the bubblers indicate iodine transport *without* being combined with cesium.

Cesium metal was added to the crucible charge because it was not possible to cast cesium in the original SRL–ANL fuel ingots. The boiling point of cesium is only slightly higher than the melting point of aluminum. Thus some loss of cesium was expected before melting. However, cesium loss continued during the experiments. Because of the significant vapor pressure of cesium, even as a solution, it was not clear whether the cesium was dissolving in the melt. Cesium was readily vaporized for irradiated uranium melts in melt refining studies.[53] Also, the release experiments at Hanford[35] indicated rapid release of cesium from aluminum fuel melts. The ORNL studies,[44,58] however, showed a more delayed release.

Deposits from the SRL–ANL thermal gradient tubes were also analyzed for tellurium in one experiment. No tellurium was found in the deposits. This is considered a significant finding of the experimental program and is consistent with previous analytical studies using SOLGASMIX.[59] These analytical calculations indicated that tellurium should be stabilized in solution by the formation of UTe_2 and possibly other tellurides. Studies of the melt refining of irradiated uranium indicated that tellurium formed rare earth tellurides, which prevented significant vaporization.[54]

No cerium or uranium was found in the SRL–ANL deposits. This finding is consistent with their very low vapor pressures. These findings suggest that little transport occurred by mechanical means (that is, there was no transport of solid oxide particles from the surfaces of the melt). Very small deposits of molybdenum were found in the deposits. Molybdenum transport could have resulted from the formation of trace quantities of volatile molybdenum oxides. Studies of the melt refining of irradiated uranium also indicated that molybdenum and cerium were not volatilized from melts.

The solubility of zirconium in molten aluminum is reported to be 0.11% at the aluminum melting point.[30] Thus it was considered likely that zirconium crucibles could be used for the SRL–ANL pool melting experiments. However, it was observed that zirconium reacts vigorously with aluminum, probably forming $ZrAl_3$. The reaction product appeared to be insoluble in the melts. Thus it was not possible to include zirconium in the original preparation of the ingots. Zirconium was thus included in the crucible charge for each experiment. No zirconium was found in the thermal gradient tube deposits of the SRL–ANL test. Zirconium is thus not expected to be readily vaporized from Al–U molten pools.

Several important observations result from the melt refining and SRL–ANL series of experiments. These observations provide the following conclusions for the behavior of the fission products:

Strontium—The experimental data indicate that the vapor pressure of strontium is significantly lower than that predicted by ideal solution behavior. This conclusion is supported by the available literature.

Barium—The experimental data indicate that the vapor pressure of barium may be greater than that predicted by ideal solution behavior. Barium release is expected to be larger as a result; however, its release is still being limited by its solubility in the alloy.

Iodine—Calculations have indicated that CsI is a likely chemical form for iodine. Experimental observations indicate that CsI is not soluble in the aluminum melts and that rapid and extensive release is expected. Other forms of metallic iodides, including UI_3, ZnI, and AlI_3, may also be expected to occur.

Cesium—The cesium release was observed to continue throughout the SRL–ANL experiments. Other data indicated that cesium is insoluble in aluminum melts and is readily vaporized. Information exists that cesium can form low volatility silicates in the melts. This may provide an explanation for the long release times found in the SRL–ANL experiments.

Tellurium—The absence of tellurium deposits in the thermal gradient tubes of the SRL–ANL experiments indicated that the tellurium may be chemically stabilized with the melt as predicted by thermodynamic analysis. Thermodynamic analysis indicates that tellurium combines with uranium to form a telluride. These results are also consistent with the results of the EBR-II melt refining experiments.

Molybdenum, cerium, uranium, and zirconium—Only traces of molybdenum were found in the SRL–ANL experimental deposits. No deposits of cerium, uranium, or zirconium were found. These results indicate that these elements are not volatile as predicted by thermodynamic analysis.

CONCLUSIONS

The information presented in this article provides the first complete macroscopic description of the phenomena and processes occurring during a melting accident in an Al–U fueled reactor. The three important phenomena describing the relocation of molten fuel have been identified. The physical parameters that influence these parameters have also been identified and experimental data used to define their influence on the relocation mechanics. This article also provides the first observations and supporting evidence that describes the influence of fission-product behavior on fuel relocation during a severe core damage accident and the influence of the oxide film of the fuel on fission-product release.

The out-of-pile and in-pile experimental programs and posttest examinations of failed metallic fuel assemblies have led to the following conclusions for irradiated fuel failure leading to melting. The undercooled fuel overheats and can fail in three distinct stages at power levels less than nominal:

- Clad blistering.
- Clad cracking.
- The fuel melts and flows through cracks in the cladding.

Clad cracking is the expected behavior for clads that melt at temperatures above the melting temperature of the fuel.

The experimental data indicate that the unirradiated metallic fuel melts and flows at a temperature near its eutectic point. The flow regime of the metallic melt is rivulet for low burnup fuel. For metallic fuel with a sufficient inventory of noncondensible gases, a fuel foam occurs, and the fuel will drain as a low-density metallic froth.

These phenomena strongly influence accident progression analysis. Rivulet flow tends to limit molten fuel motion under conditions of localized burnout. Localized burnouts can remain stable and not propagate if small enough; the molten mass remains in the vapor-blanketed regions. This condition can remain stable for several hours. This condition is similar to the incident that occurred in one of the SRS melting incidents.

The molten fuel mass will span the coolant channel and contact the surrounding fuel element before relocating if the coolant channel thickness is less than the critical thickness for relocation of the rivulet. This can lead to a coolable geometry if the surrounding elements have adequate cooling. In contrast, if the surrounding elements are not sufficiently cooled, the molten fuel mass and the surrounding melting element may intermix. This intermixing requires that the effects of the oxide film and surface tension be overcome.

Burnup tends to increase the interaction between failing fuel elements by the process of foaming. The agglomeration of fission-gas bubbles during the rapid melting of these metallic fuels tends to result in rapid swelling near the melting point of the fuel alloy. This swelling tends to reduce the coolant channel thickness. This phenomena can promote melt propagation to surrounding elements.

The following observations summarize the cermet fuel-melt behavior:

- The irradiated cermet fuels do not foam.
- The irradiated cermet fuel exhibits extensive blistering at the fuel-cladding interface.
- The unirradiated fuel shows little evidence of the gross blistering as was observed for the irradiated cermet fuel.
- The melt failure of the unirradiated cermet fuel is defined in terms of a melt slumping of the aluminum matrix.
- The irradiated cermet fuel failure sharply contrasts with that of the unirradiated cermet fuel in that only the clad is observed to slump away from the fuel meat. The irradiated cermet fuel meat remains a rigid body above approximately 1200 °C.
- The U_3O_8 particulate appears to oxidize the aluminum matrix during irradiation, which results in a significant increase in the melting temperature.

Fission-product release is a key mechanism in understanding the fuel foaming process and the behavior of the molten fuel pools formed in metallic Al–U fueled reactor accidents. The fuel swelling process in conjunction with the breakup of the oxide film introduces stochastic behavior into the fission-product release process. The fuel foaming–swelling process in conjunction with the stochastic nature of the oxide film rupture introduces scatter into the fission-product release data base. These effects are, in part, associated with the amount of fuel burnup (i.e., volatile fission-gas inventory).

New information has been provided to understand fission-product chemistry during severe accidents. The significant findings are that tellurium remains in solution in the alloy and is not readily vaporized from the melts. Strontium is released at rates less than predicted from ideal solution behavior. Barium, however, is released at rates slightly higher than predicted by ideal solution behavior. The behavior of the other fission products is as expected from ideal thermodynamic analysis. The behavior of iodine in the melts indicates that it is readily released from the alloy as expected.

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R&D Activities on Safety Aspects of Future PWR Plants Performed at KfK

By B. Kuczera[a]

Abstract: *In the present discussion on the next generation of light-water reactors (LWRs), two conceptional tendencies are discernible, depending on the point of view adopted: a revolutionary tendency, which is based above all on passive and inherent safety features, and an evolutionary tendency, which relies on the existing commercially proven LWR technology and operation experiences. In line with the latter trend, particular considerations are being made on containment concepts for future pressurized-water reactors (PWRs), which are accompanied by appropriate research and development studies on severe accident containment loadings. These include the estimation of loads that might result from an energetic in-vessel steam explosion, high-pressure failure of the reactor pressure vessel, or dynamic hydrogen combustions. First estimates on corresponding upper limits of loading are presented. Complementary investigations concentrate on the long-term reliable removal of the decay heat from the core melt and from the accident atmosphere in the containment, respectively. In this context two core-melt cooling concepts (core catchers) are presented that can serve as innovative elements in future PWR plants. The general goal of these studies is to contribute, from the technical point of view, to the development of an advanced containment that allows significant radiological consequences to be excluded for the environment under severe reactor accident conditions.*

The light-water-reactor (LWR) equipped nuclear power plants presently operated in the western hemisphere rely on a safety concept that was developed in the 1970s. The overall objective of reactor safety is to protect the population against dangerous releases of radioactive materials. The corresponding strategy—well known as "defense-in-depth"—is based on a multiple confinement of these materials by several sequentially arranged physical barriers and on a multilevel protection system to ensure continued integrity of these barriers. In this context, the ultimate barrier is the containment of a reactor plant, which may be illustrated by the example of a modern 1 300-MW(e) pressurized-water reactor (PWR) of the German CONVOY series. Figure 1 shows a CONVOY reactor plant with the essential safety installations and its containment system, which encloses the primary coolant circuit. The large dry containment consists of a spherical steel vessel that is 38 mm in wall thickness and 56 m in diameter. The free containment volume is about 70 000 m^3. The 0.6-MPa design pressure is determined from the enthalpy of the primary circuit; in a loss-of-coolant accident (LOCA), it is released into the containment. The containment is protected from external impacts by a 1.8-m-thick concrete structure. The annulus between the concrete structure and the containment shell is permanently exhausted via a filter section; in case of an accident, the exhaust flow is directed through a special accident filter device that largely excludes the radioactive materials leaking from the containment that would be directly released into the environment. This is the present situation.

In the discussion about the next generation of LWRs, the defense-in-depth concept continues to be the fundamental means of ensuring the safety of nuclear plants. In this context, however, there are two tendencies to recognize, depending on different points of view adopted: (1) a revolutionary tendency, which is based above all on passive and inherent safety features; and (2) an evolutionary tendency, which relies on the existing commercially proven LWR technology.[1] Whereas the revolutionary approaches are directed to novel reactor systems in a technological virgin country, the evolutionary tendencies are concentrating on technical improvements of the already proven technology with a goal of further reducing the residual risk perceived by the public to result from the operation of nuclear power plants. This article is mainly oriented toward the latter tendency, even though various features treated can be considered to be generic.

With respect to technological advances, the first question is in which fields will improvements be most effective? The definition of risk as the product of the frequency of occurrence of an event and the scope of damage points in two directions: improving accident prevention by additional preventive measures and/or

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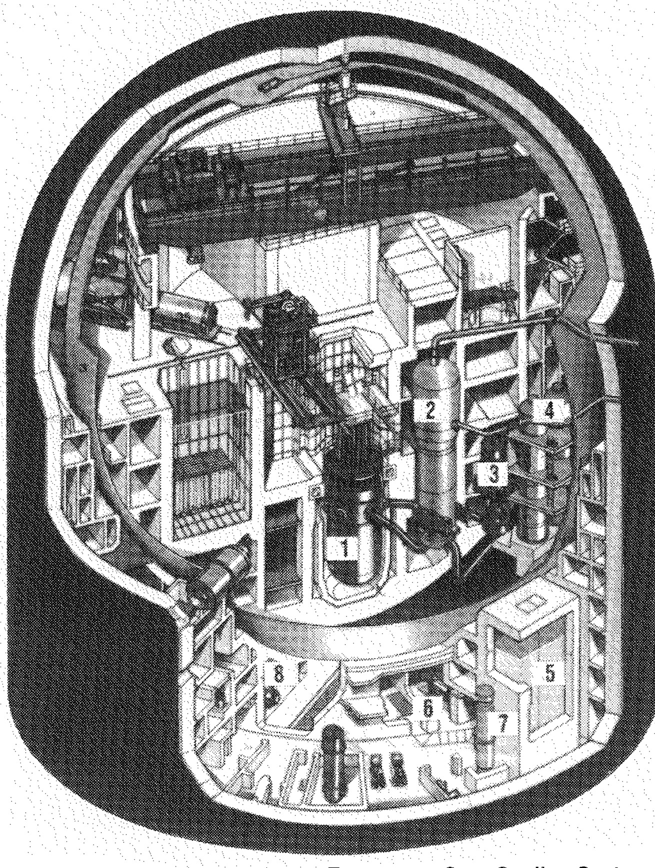


Primary Loop
(1) Reactor pressure vessel
(2) Steam generator (4)
(3) Reactor coolant pump (4)

Emergency Core Cooling System
(4) Accumulator (4x2)
(5) Flooding reservoir (4)
(6) Safety injection pump (4)
(7) Residual heat exchanger (4)
(8) Residual heat removal pump (4)

Fig. 1 Safety systems in a 1300-MW(e) pressurized-water reactor (CONVOY series). (Source: Siemens/KWU, Erlangen.)

limiting the consequences of an accident by mitigative measures (see Fig. 2). A decisive improvement in accident prevention by modifications in the design (e.g., by increasing the number of redundancies) meets with fundamental difficulties, considering the high level achieved already. Obviously, optimum engineering solutions have already been applied in the majority of cases. So, should a major improvement be achieved, the consequences of severe accidents must be further mitigated (that is, in the extreme case of a core meltdown accident, the majority of radioactive materials must be retained within the plant). This would avoid, even in this limited case, the necessity of an evacuation of the nearby population, and the contamination of large land surfaces over extended periods

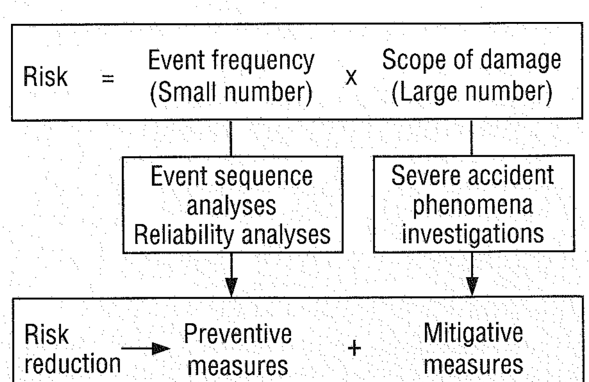


Fig. 2 Approaches to reducing risk.

could be ruled out. This consequently means that the effectiveness of the containment must be improved.[2]

In regard to extreme accident-induced containment loadings, the German Risk Study on Nuclear Power Plants–Phase B (GRS–B)[3] points to a number of phenomena whose consequences—albeit with very little likelihood—might jeopardize the integrity of the present PWR containments. These include the high-energy steam explosion in the reactor pressure vessel (RPV), failure caused by melt-through of the RPV at high primary circuit pressure, a dynamic hydrogen deflagration–detonation a few hours after core meltdown, and long-term erosion of the basemat and basemat penetration resulting from core melt–concrete interaction. With these phenomena in mind, questions are frequently asked about the appearance of a containment that is given an innovative design such that it withstands the effects of accidents, including those which cannot be controlled by the safety systems installed. At KfK the number of relevant studies has significantly increased in recent years.[4] In the underlying work, there are fewer studies on the design of a new containment in terms of construction measures; this will remain a task tailored to the power plant manufacturing industry rather than to the theoretical and experimental validation of realistic upper limits of containment loadings. First results have been published recently.[5–7] In the following sections, an attempt is made to illustrate the present status and the perspectives revealed in the KfK studies.

HIGH-ENERGY ACCIDENT PHENOMENA

In this section investigations of in-vessel steam explosions, the high-pressure (HP) path, and hydrogen combustion events will be presented. The various approaches chosen involve the development of best estimates of energy releases associated with these phenomena. A modern 1300-MW(e) unit of the CONVOY series will serve as the reference plant.

Energetic Steam Explosion

In GRS-B a cautious estimate was made of the energy conversions that potentially take place in a steam explosion supposed to occur in the RPV. The assumption was made that after a LOCA that cannot be controlled a core-melt mixture (corium) of 10 t with a thermal energy of 15 GJ penetrates in a continuous flow into the lower plenum of the RPV and instantaneously reacts with the water. With the further assumption of an (thermal-to-mechanical) energy conversion factor of 0.1, the conclusion was that an in-vessel steam explosion accompanied by mechanical energy release of more than 1.5 GJ is a very low probability. Supplemental strength analyses have shown that the RPV withstands the resulting shock-wave-induced loadings so that the integrity of the steel containment is not endangered (i.e., an α-mode containment failure was not considered).

As evident from the discussion in scientific publications, however, the opinions on this item are not uniform. In critical comments, quite higher mechanical energy releases than those just indicated are reported. In this respect, we do not think that we will reach a new order of magnitude but deem an uncertainty factor of 2 appropriate for the present investigations; from a scientific point of view, this may appear somewhat arbitrary, but, from an engineering viewpoint, this does not seem to be unreasonable. In this way a maximum release of energy of 3 GJ (instead of the 1.5 GJ) is obtained as an approximate figure.

In regard to the consequences in the RPV of such an amount of energy released, we adopt a pessimistic scenario, which was developed some years ago by Theofanous et al.[8] The underlying sequence is illustrated in Fig. 3. The explosion energy of 3 GJ causes an RPV rupture in the lower head region with the assumption that, by mechanical deformations, accelerations of masses, and steam expansion, roughly two-thirds of the energy is dissipated. It is further assumed that within the RPV a corium slug is accelerated upward, which accumulates about 700 MJ of kinetic energy that, finally, is partly consumed in the upper part of the RPV through deformation of the RPV internal structures and excessive loading of the bolts of the vessel head. The sketch on the right side of the figure shows that, ultimately, the broken-off RPV head with a residual kinetic energy portion of about 150 MJ might greatly endanger the integrity of the containment. This is the general scenario as described by Theofanous. If actual CONVOY conditions are considered, we have, however, a good chance of demonstrating that, most importantly, the bolts of the vessel head will not fail, so the "danger due to missiles" does not have to be considered.

On the basis of this background, our research and development (R&D) work on molten fuel–coolant interaction (MFCI) is concentrating on three key phenomena that *inter alia* essentially determine its destructive potential.[9]

1. First, we try to demonstrate that the molten corium mass penetrating simultaneously into the lower RPV plenum and interacting with water in the "premixing phase" can be limited to about 10 t. In this context the

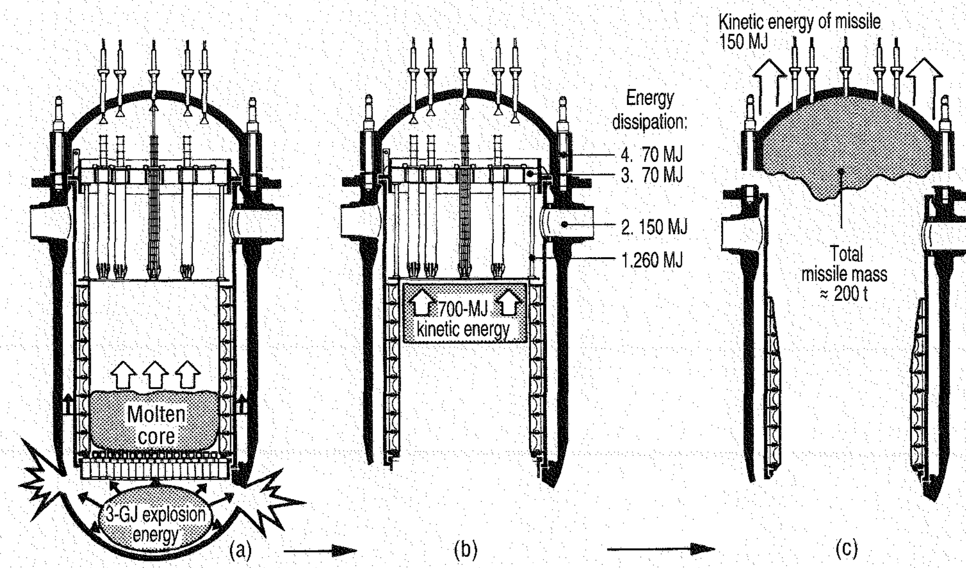


Fig. 3 Energy partition associated with in-vessel steam explosion scenario.

heat transfer from the melt to the coolant (by radiation and evaporation) exerts a dominating influence on the displacement of water out of the reaction zone, which, finally, may result in an autocatalytic self-limitation of the MFCI. Dedicated single-effect tests on heat transfer in which hot (T = up to 2500 °C) metal spheres drop into a water pool are to provide a general data base allowing existing three-dimensional (3-D) models on steam explosion to be verified.[10]

2. The second aspect concerns the conversion of thermal energy into mechanical energy. In the majority of experiments[11] performed so far, the energy conversion ratios range from 0 to 0.03 with an uncertainty factor of 2. In a number of experiments involving (5 to 40 kg) thermite melts and water, we try to find out whether an upper limit of 0.15 can be set to the range of conversion. Figure 4 shows sketches of the experimental facilities under construction for these investigations.[12]

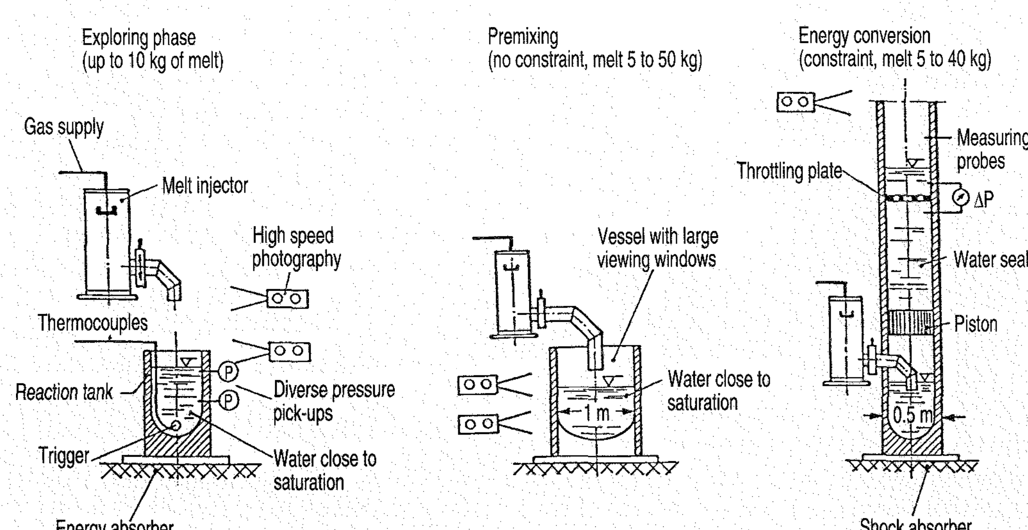


Fig. 4 Various small- and medium-scale experiments on in-vessel steam explosion phenomena.

3. The third question concerns the dissipation of energy through plastic deformation of the RPV internal structures. We think that it can be demonstrated in experiments that a considerable portion of the kinetic energy of the molten core slug can be absorbed through plastic deformation. The situation in the RPV has been illustrated once more in Fig. 5, left-hand side; the scaled-down (1:10) experimental equipment is represented on the right. In the experiments corium is simulated by a slug of 80-kg liquid lead alloy (with a low melting point), which is first accelerated from below to a velocity of about 130 m/s and then penetrates into the upper in-vessel core structures. From the size of the impulse hitting the upper vessel head, conclusions can be drawn on the energy of deformation consumed.[13]

Melt-Through of the RPV at High Primary Circuit Pressure

The analyses performed within the framework of GRS-B reveal that about 98% of all core meltdown accidents are expected to occur at high primary circuit pressure (HP). Accident management measures have not been considered. When included, such measures will reduce the frequency of occurrence of HP scenarios to the order of 10^{-7}/yr. Although these are sequences of events having an extremely low likelihood of occurrence, they must be included in the discussions about the safety of future reactor plants.

One can imagine that in an HP core meltdown accident, after dryout of the core zone, the melt relocates into the lower head region of the RPV, and the RPV fails because of thermal overload with a circumferential rupture of the lower vessel head. (Note: The CONVOY RPV has no penetrations in the lower head.) Depending on the size of the rupture opening, the resulting pressure relief of the primary circuit gives rise to reaction forces acting on the RPV; these forces might destroy the RPV support system. The first calculations using the RELAP5 code give an impression of the shock forces acting on the RPV.[14] As shown in Fig. 6, the forces vary between 200 and 300 MN with the size of the rupture cross section; they are active for about 50 ms. In the vicinity of the RPV and in the reactor pit, respectively, the pressure rise, which temporarily might attain values up to 25 bar, must be considered as closely correlated to the depressurization events.

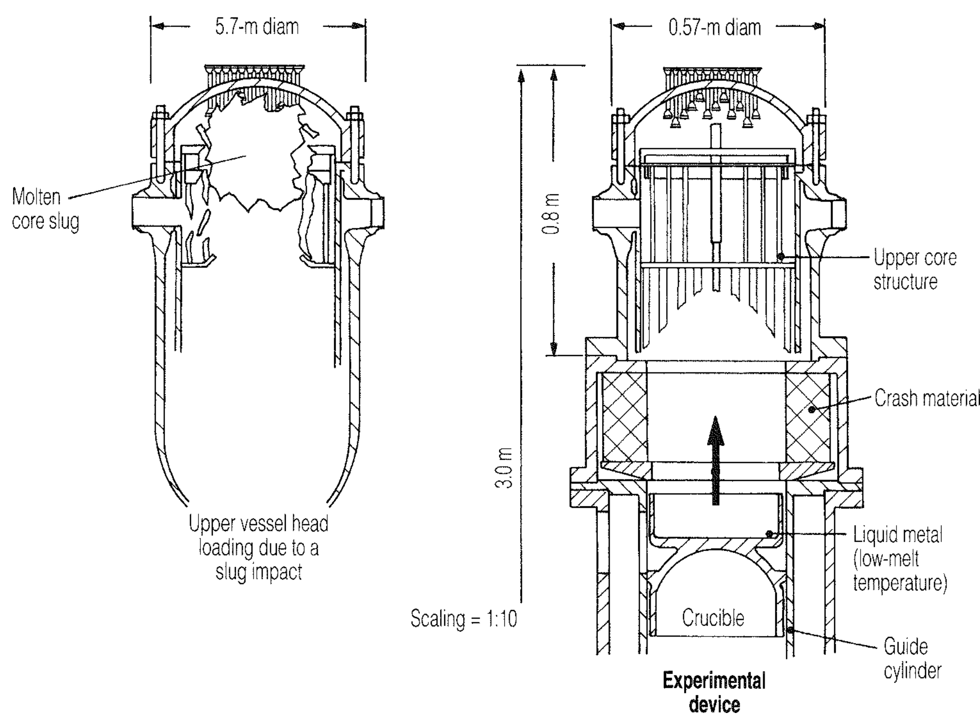


Fig. 5 Experimental investigation on energy dissipation by plastic deformation of vessel internal structures (sketch of the BERDA facility).

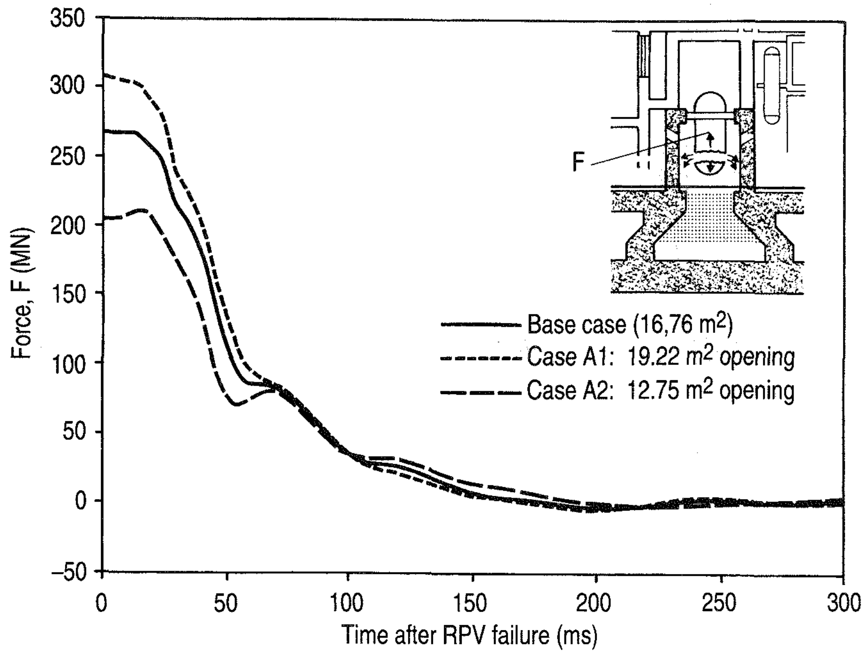


Fig. 6 Thrust load on the reactor pressure vessel in case of high-pressure lower head failure.

The future work in this sector will be devoted to a refinement of analysis of these transient phenomena. The theoretical activities on fracture mechanics of the RPV lower head will be in the foreground; these are to provide indications regarding the type of failure and the development vs time of the rupture opening of the vessel. Experimental investigations that cover the temperature range up to 1000 °C are in progress on the strength properties of the RPV material. In the analysis of the thermodynamic processes, the extent to which two-dimensional modeling of the outflow events immediately after the vessel failure will be necessary to validate the results obtained is examined. In this context, also the question of the extent to which phenomena related to the "direct containment heating" [that is, fast exothermal oxidation of the zirconium particles (from the ejected core-melt mixture) in the steam atmosphere] should be included in the considerations of the HP scenario must be examined.

Hydrogen Combustion Processes

In a core meltdown accident, two phases occur during which considerable amounts of hydrogen (gaseous H_2) are generated and released into the containment atmosphere. In the early phase of core degradation, this is caused by exothermal oxidation of Zircaloy cladding tubes (zirconium–vapor reaction), and, in a later phase, after RPV failure, by oxidation of the metal fractions of corium during melt–concrete interaction.[15,16] Both processes may cause the H_2 mass present in the containment to attain 1300 to 1700 kg. However, if further metal oxidations are included in the considerations, a conservative upper limit of 2000 kg of H_2 might be conceivable.

For some time various preventive measures that are intended to avoid a risk to the containment integrity resulting from H_2-combustion processes accompanied by detonations have been discussed. In Germany, the so-called dual concept, in which the H_2-concentration in the containment should be reduced at an early stage, is the most elaborated one.[17] The concept provides for catalytic H_2-recombination on specially conditioned foil surfaces and (as a diverse measure) H_2-deflagrations initiated by battery-fed spark igniters. However, there are some doubts about the extent to which these safety-related measures will reliably exclude the occurrence of an H_2-detonation. Therefore the question of the hazard potential on the containment integrity that might result from an (deliberate or random) H_2-ignition through deflagration or detonation has been repeatedly asked in a critical approach.

The parameter studies related to deflagration started from a homogeneous distribution of H_2 in the containment volume. In case of complete combustion under adiabatic isochoric conditions (AICC = adiabatic, isochoric, and complete combustion), the peak pressures and temperatures represented in Fig. 7 as a function of the hydrogen/oxygen ratio Φ are obtained (for explanation: $\Phi = 1 \Rightarrow$ stoichiometric conditions; $\Phi < 1 \Rightarrow$ oxygen in excess; $\Phi > 1 \Rightarrow H_2$ in excess in the containment atmosphere). For the case under scrutiny of 2 000 kg of H_2 in 70 000 m^3 volume ($\Phi = 0.82$), a combustion pressure $P_{AICC} \approx 1.5$ MPa and a combustion temperature $T_{AICC} \approx 1\,280$ K are obtained. Even if sufficient H_2 were present to burn all the oxygen (2 400 kg of H_2, $\Phi = 1$), a peak pressure of $P_{AICC} \approx 1.7$ MPa ($T_{AICC} \approx 1\,370$ K) would be obtained; this means that the present uncertainties concerning the assumed amount of H_2 present have little effect on the maximum combustion pressure.[16] The indicated temperatures drop at a relatively fast rate (within about 15 minutes) to a level of about 200 °C.

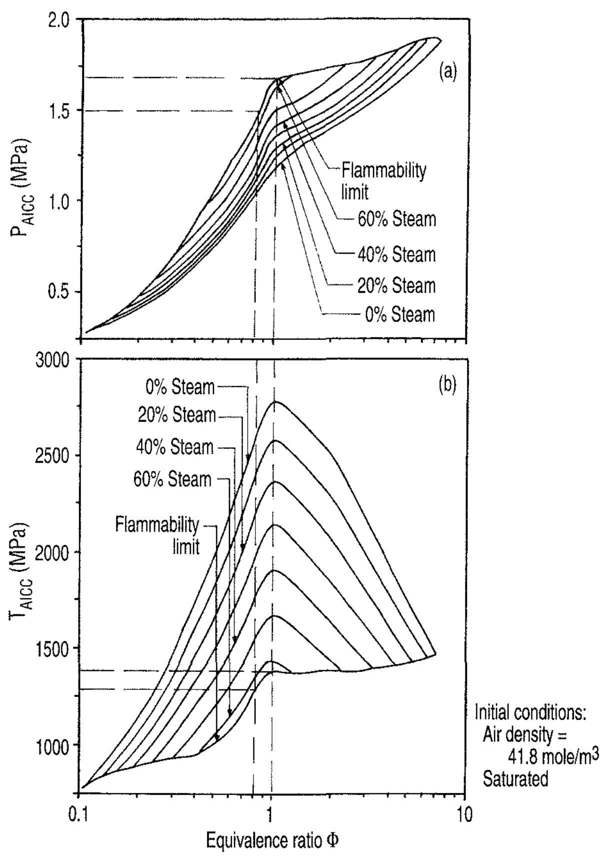
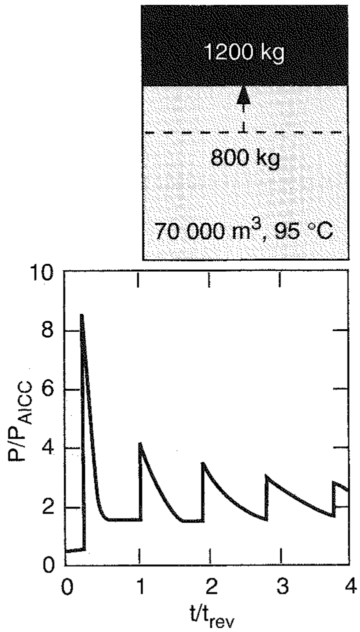


Fig. 7 Pressure (a) and temperature (b) plots from slow hydrogen deflagrations in containment. AICC = adiabatic, isochoric, and complete combustion.

If an enrichment in H_2-concentration in the upper part of the containment dome is assumed, instead of the homogeneous distribution of H_2 (2 000 kg of H_2 in a 70 000-m^3 air–steam mixture roughly corresponds to an H_2 fraction of 20%), the following conservative scenario can be formulated.[18] As outlined in Fig. 8, 1 200 kg of H_2 is assumed to be present in the dome and 800 kg of H_2 in the lower part of the containment. In a fire developing in the bottom part, the unburned mixture in the dome is supposed to become compressed, and, for example, by free jet ignition, this would cause a detonation. One-dimensional computations provide the pressure–time curve traced in Fig. 8, with the pressure axis normalized at $P_{AICC} = 1.18$ MPa and the time axis at the time of shock-wave travel in the burned gas mixture ($t_{rev} = 29.1$ ms). Accordingly, a peak pressure of 10.5 MPa (105 bar) is attained in the first reflected detonation wave, and an impulse of about 0.06 MPa • s is imparted to the containment on a surface of about 1 400 m^2. We are aware that these results are preliminary and might indicate, conservatively, maximum load conditions. In the meantime, more refined 3-D detonation analyses are under way.

The upper limits of containment loadings resulting from hydrogen combustion events can be indicated in quantitative terms to be in the range of 1.5 to 1.7 MPa static pressure and a transient pressure impulse of 0.06 MPa • s (acting on about 1 400 m^2 of surface area).

Future activities will also elaborate on contributions directed toward the German qualification of the dual concept (e.g., H_2-distribution and arrangement in space of the igniters) and concentrate on the development of realistic multidimensional models of computation.[19] The improved detonation models are to allow the containment internal structures as well as their feedback on the combustion processes (e.g., flame acceleration by generation of flow turbulences) to be included in an adequate manner. Simultaneously, detonation experiments are being performed in tube geometry (43 cm in diameter and up to 12 m in length) to investigate the dominant physical parameters; the experimental facility is shown in Fig. 9. Further large-scale tests in combined compartment and sphere geometry (16 m in diameter) are being prepared in close cooperation with external institutions. These activities will contribute to the extension of the available data base, which is needed for an adequate model verification, and validation, and, finally, to the support of a more prototypic simulation of hydrogen combustion phenomena in a containment.



- Scenario is a combination of limiting parameter values:
  - 2000 kg H_2 released
  - Distribution ≙ test HDR T31.5: 1200 kg in dome, 800 kg in lower rooms
  - Temperature 95 °C (much steam)
  - 800-kg burn, compress dome gas
  - 1200-kg detonate
  - 1-D-planar wave
  - Normal reflection at ceiling
- Global detonation load
  - Peak pressure 10.5 MPa (P_{AICC} = 1.18 MPa)
  - Reflected detonation impulse 0.06 MPa•s (up to t/t_{rev} = 1, t_{rev} = 29.1 ms)
  - 1400 m^2
- More realistic 3-D detonation model is needed to reduce conservatism. Lower global loads are expected.

Fig. 8 Conservative hydrogen detonation scenario.

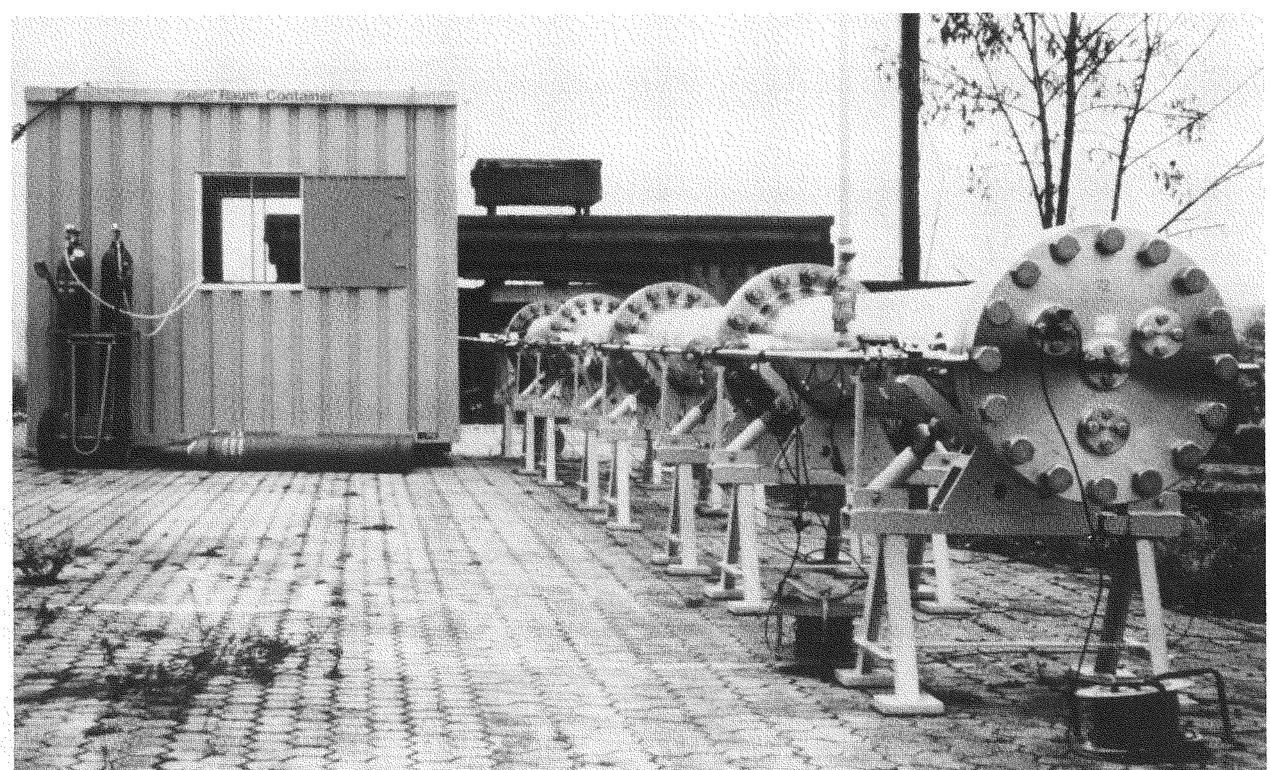


Fig. 9 Hydrogen detonation tube test facility.

CONCEPTIONAL CONSIDERATIONS ON STRENGTHENING THE CONTAINMENT FUNCTION

As stated at the beginning of this article, the design features of a future PWR containment are not in the foreground of these R&D activities. However, it is often necessary to make design-oriented considerations to be able to define modified assumptions to carry on the work. In this sense, for instance, the containment represented in Fig. 10 has been conceived in cooperation with the Institut fur Massivbau und Baustofftechnologie of Karlsruhe University.[4,5] The containment geometry is cylindrical in the bottom part (about 65 m in diameter) and becomes hemispherical in the upper part. In principle, the containment is a composite containment, which means an inner tight steel shell of about 38 mm in wall thickness surrounded by an approximately 2-m-thick reinforced-concrete wall. The steel shell may be designed to withstand conditions similar to those prevailing in present containments (e.g., 0.6 MPa/145 °C) because it is supposed to absorb only a minor fraction of the maximum static loading of about 1.5 MPa. As shown in the A-A sectional view of Fig. 10, the annulus between the two shells is bridged by double-T-beams. In case of heavy loading of the inner containment (e.g., by hydrogen combustions), the steel shell will expand, and via the T-beams frictional connection will be established with the concrete wall, which then absorbs the residual load and thus prevents overload-induced failure of the steel shell. At the same time, a type of chimney is built up in the annulus that supports natural draft cooling of the steel shell and hence passive decay heat removal from the containment. This aspect will be treated in more detail later in this article.

The R&D studies on phenomena associated with severe accidents are not focused on the concept outlined previously but are generic in nature, covering wide ranges, so the results obtained will be of interest to other future containment concepts as well. For instance, we are taking great interest in development trends in industry. The conceptual design considerations relating to a European Pressurized Water Reactor (EPR)[20] should be mentioned first in this context. Although the respective phase has not yet been terminated, clear contours of this future project are already visible in the representation in Fig. 11, and we think that our R&D results will support the trend of innovation in the future elaboration of the project.

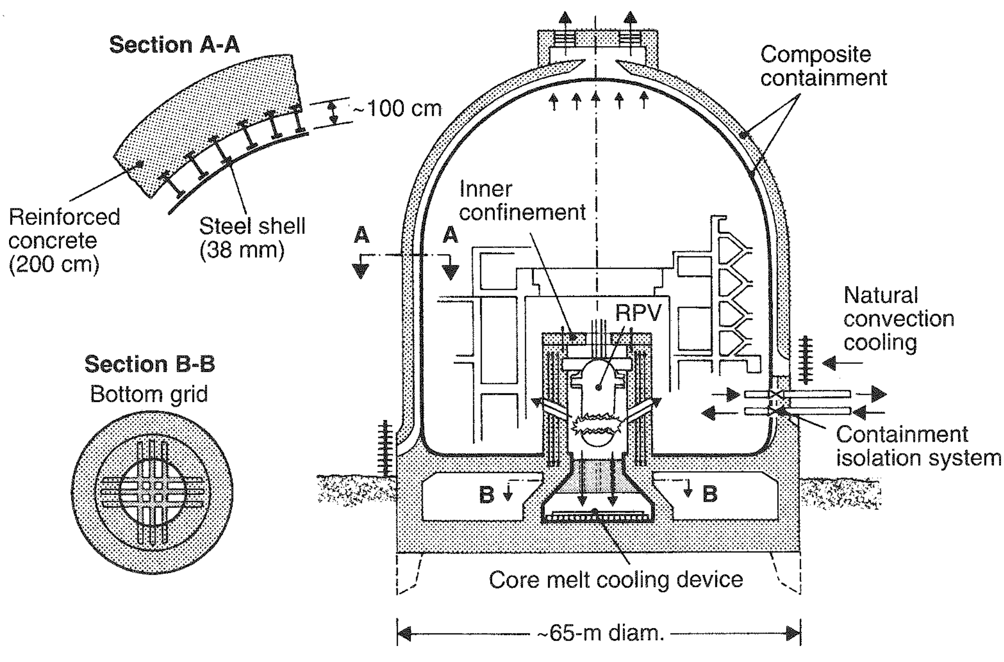


Fig. 10 Conceptual design of a composite pressurized-water-reactor containment. An inner tight steel shell is surrounded by a strong outer reinforced-concrete structure (Source: J. Eibl, Karlsruhe University).

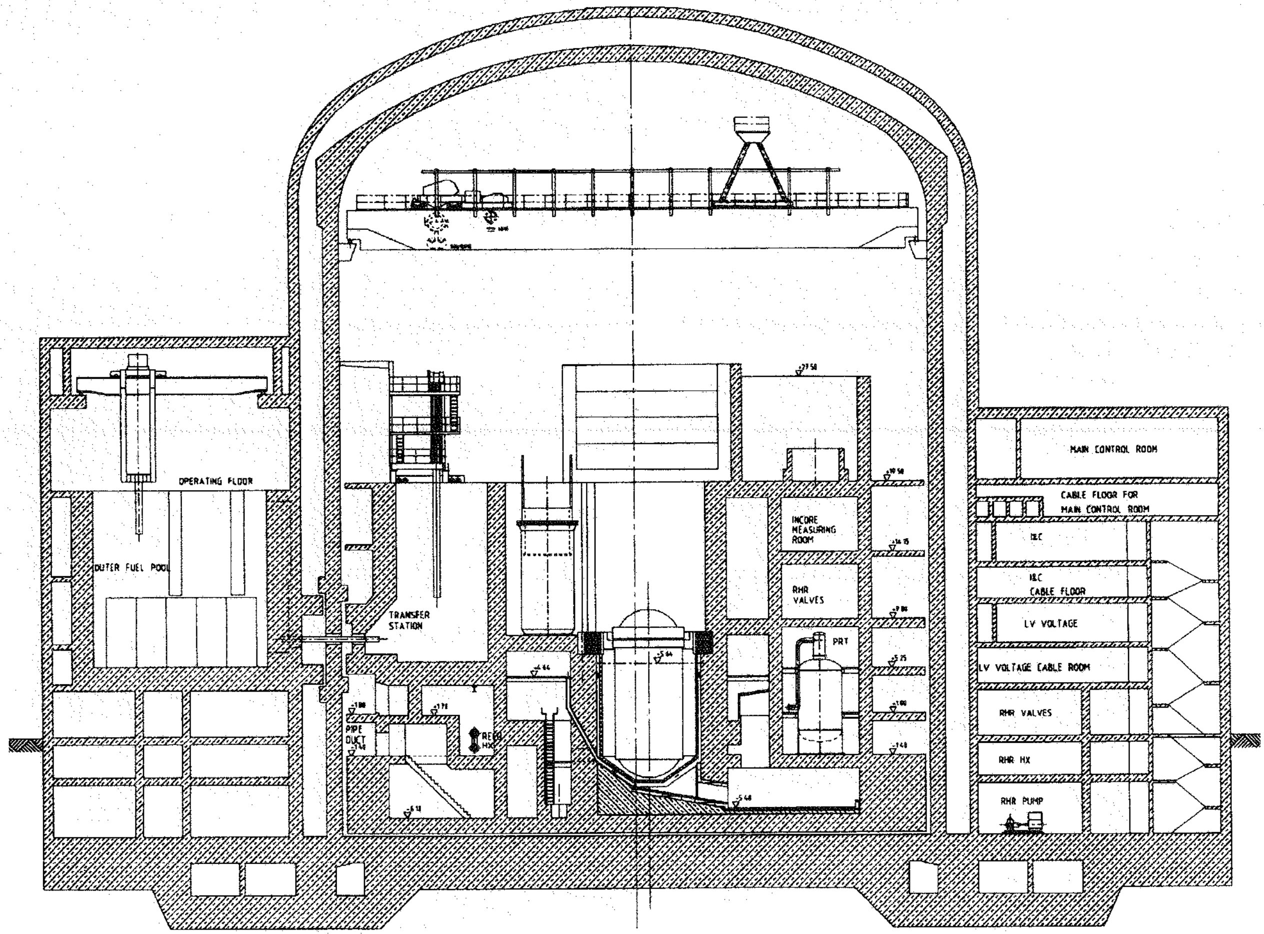


Fig. 11 Concept of the European pressurized-water-reactor containment. (Source: Nuclear Power International, Paris).

CORE-MELT COOLING SYSTEMS

The requirement of containment integrity also comprises the retention of the core melt in the containment. For a 1300-MW(e) PWR, we assume that the corium mass amounts to about 200 t. One hour after reactor shutdown the decay heat power is about 1% of the thermal reactor power, which corresponds to 37 MW(th), and it drops within 10 days to about 7 to 8 MW(th). To meet the requirement, two demands must be satisfied: first, long-term cooling of the core melt in the containment must be guaranteed to prevent molten core–concrete interaction (MCCI) (that is, reactor basemat erosion) from taking place; second, adequate decay heat removal from the containment must be ensured, if possible by passive mechanisms, to avoid long-term containment overpressurization.

Cooling of the melt implies a problem of heat removal from a volume with internal heat sources. The ratio of volume to surface plays a crucial role. The larger the surface of a given volume, the more effective is the heat removal. With this objective in mind, two concepts on core-melt cooling devices were developed and will be discussed briefly.

Under the first concept, the aim is to achieve the largest possible corium surface by plane spreading and fragmentation of the melt;[21] Fig. 12 shows an engineering proposal. The so-called core catcher is placed in an enlarged reactor pit, which toward the top is protected against mechanical loadings by a massive energy absorbing grid (discussed previously). The core catcher consists of a perforated steel plate supported by a special structure. The space beneath the plate is filled with sump water. The hollow plugs that are welded in are initially

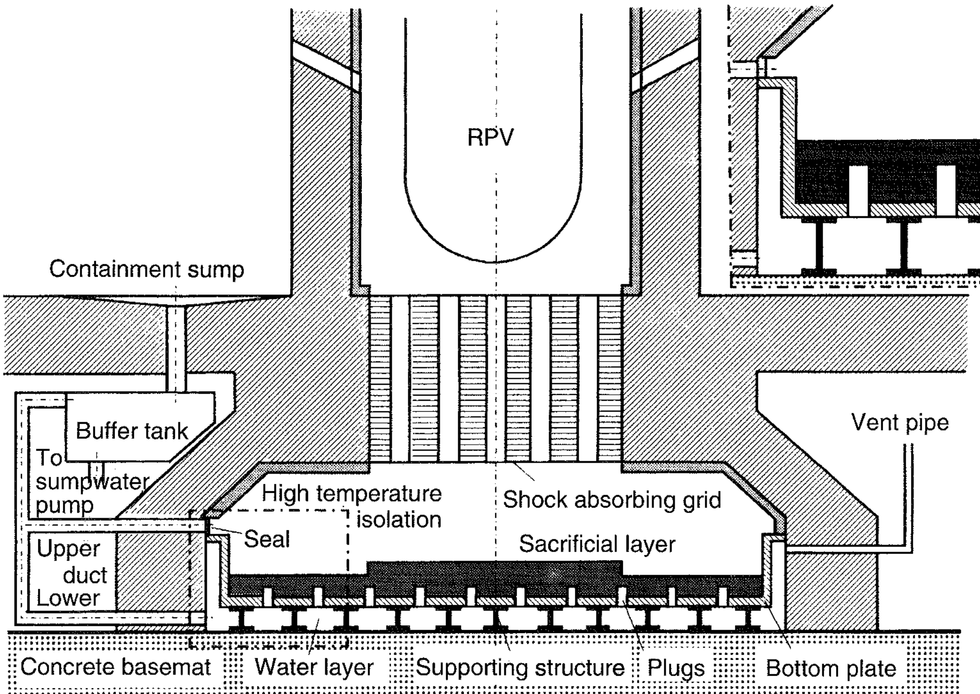


Fig. 12 Reactor building of the European Pressurized Water Reactor (EPR). An inner prestressed-concrete containment is surrounded by a second reinforced-concrete wall (Source: Nuclear Power International, Paris).

closed and prevent the reactor cavity from being flooded precociously. Upon impingement the melt will start to interact with the sacrificial layer. With the material of the sacrificial layer appropriately selected, the horizontal spreading of the melt can be promoted by reduction of the viscosity of the corium. With progressing interactions an increasing number of hollow plugs are destroyed, and sump water contacts the layer of the melt from below, penetrates it, and finally floods the major part of the reactor pit. In this way a porous stratified corium bed is produced from which the decay heat is essentially removed by sump-water evaporation. The first simulation experiments in which a thermite melt was flooded with water injected from the bottom have confirmed the efficiency of the concept of decay heat removal. Preparatory work is being done to validate the concept on the basis of large-scale experiments with induction-heated corium simulants. The BETA facility[22] used in the past for comprehensive MCCI experimental studies is being modified. The new facility, shown in Fig. 13, will allow planar melt spreading and melt fragmentation phenomena to be investigated under more representative boundary conditions.

The second core catcher version is presented in Fig. 14.[21] In this version the goal is 3-D core melt spreading in the reactor cavity. This is accomplished by completely filling the reactor cavity with a particle bed (diameter of spheres about 5 cm) with staggered catching pans and deflection plates integrated in the bed. To avoid chemical reactions with the corium, only oxide ceramic materials are used. The top layer of the bed consists of particles that have a high melting point (MgO and ZrO_2) and is not flooded to avoid steam explosion phenomena. The major bottom part consists of an Al_2O_3 bed and is flooded with sump water. This version is effective when the melt enters the particle bed from the top and spreads successively on the vertically arranged pans by overflowing. Simulation experiments involving thermite melts have confirmed the mechanism of propagation in a dry configuration; complementary experiments in a flooded geometry are being prepared.

Both proposals are based on the same principle of decay heat removal: cooling of the melt by flooding and sump-water evaporation into the containment with the reflowing condensed steam from the inner containment surface establishing a self-sustaining steam–water circulation.

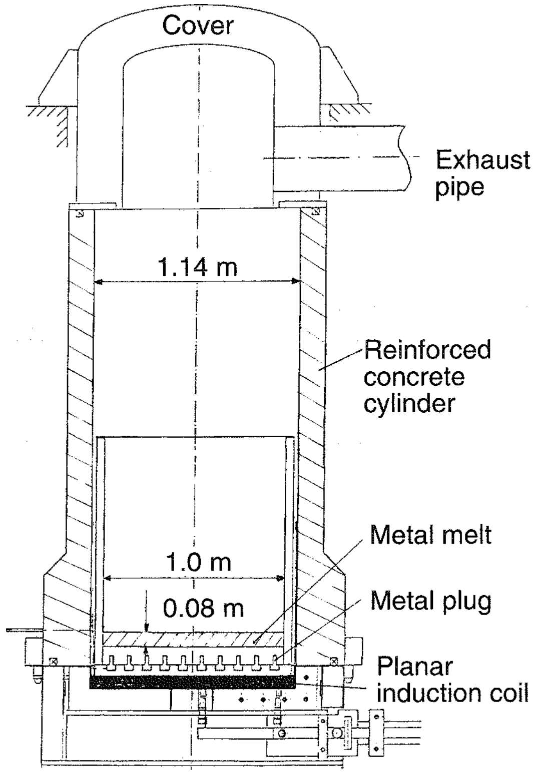


Fig. 13 Modified BETA facility for planar melt fragmentation and cooling experiments with sustained inductive melt heating.

Supplementing this description, it should be pointed out that further concepts on core-melt cooling are being developed elsewhere.[20,23] However, all concepts are not yet sufficiently validated by experiments for an evaluation based on comparison to be made.

DECAY HEAT REMOVAL FROM THE CONTAINMENT

Before dealing with innovative means of decay heat removal from the containment, it would be beneficial to look at a similar PWR accident scenario described in GRS–B. In the low-pressure core meltdown accident (LP-path) considered there, the failure of the active residual heat removal systems for decay heat removal from the containment is postulated. The pressure development in the containment to be expected from that accident scenario is shown in Fig. 15.[24] According to this figure, the pressure rises to the design pressure of the containment (0.6 MPa) within about 3 days, and after another 2 days it would attain the level of failure (0.9 MPa). Meanwhile, management measures have been taken to counteract this development accident by installing a filtered venting system for the containment (Fig. 16) that allows timely pressure relief of the containment.[25] A question that is asked frequently, with a view to future PWR plants, is how the pressure rise outlined previously would propagate in a containment reinforced by construction measures

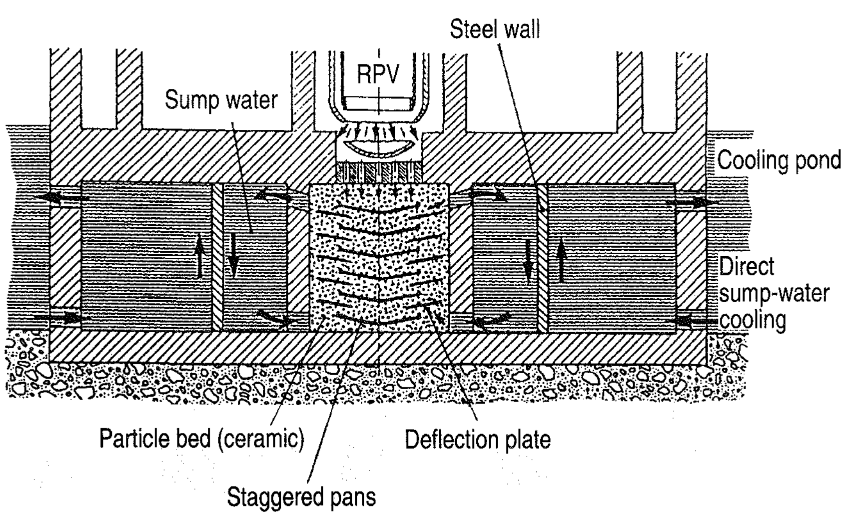


Fig. 14. Core catcher consisting of staggered ceramic plate configuration in particle bed with integrated passive sump-water cooling.

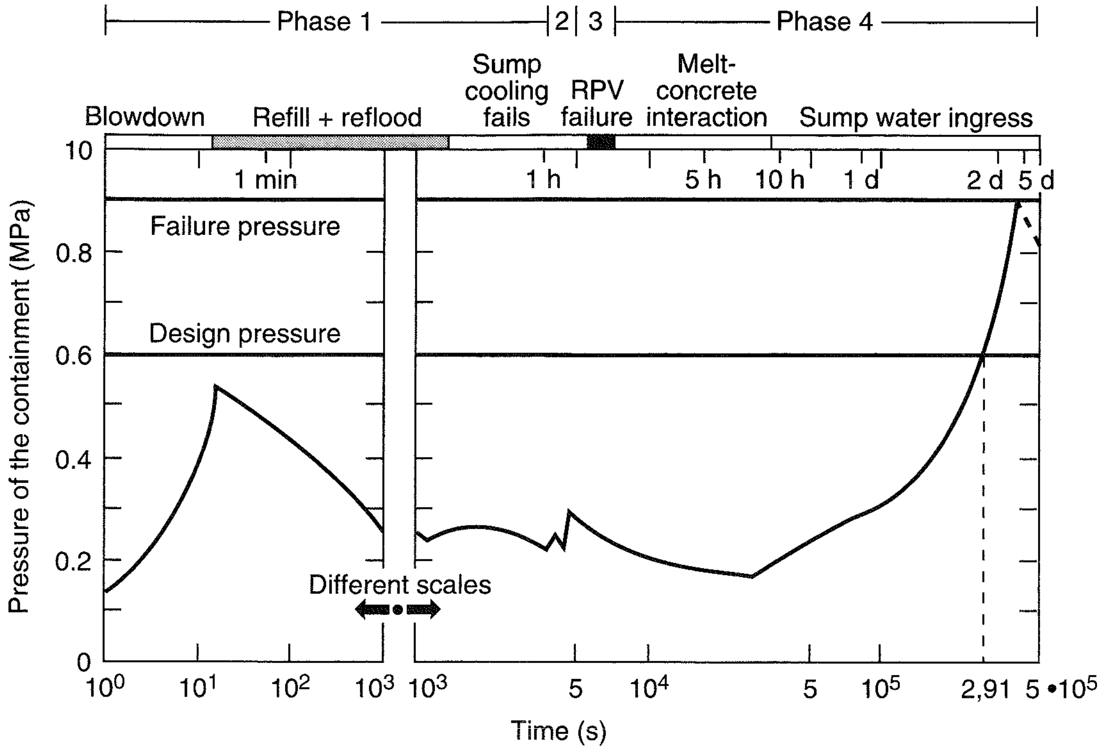


Fig. 15 Pressurized-water-reactor containment pressurization during core-melt accident (LP-path).

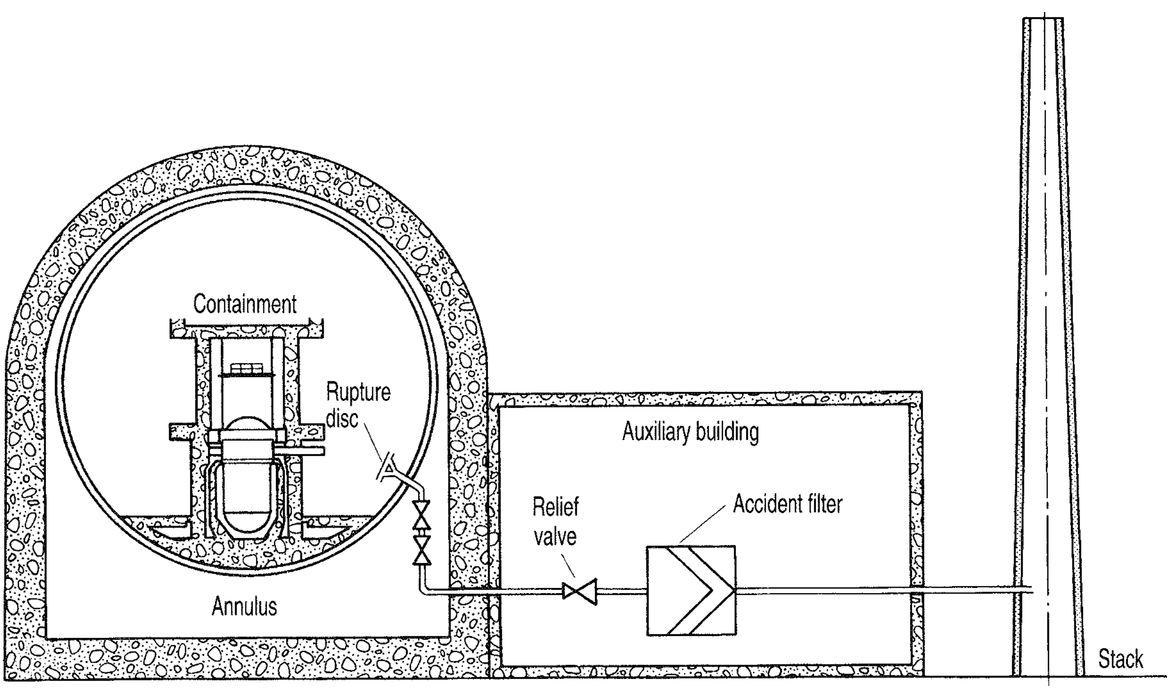
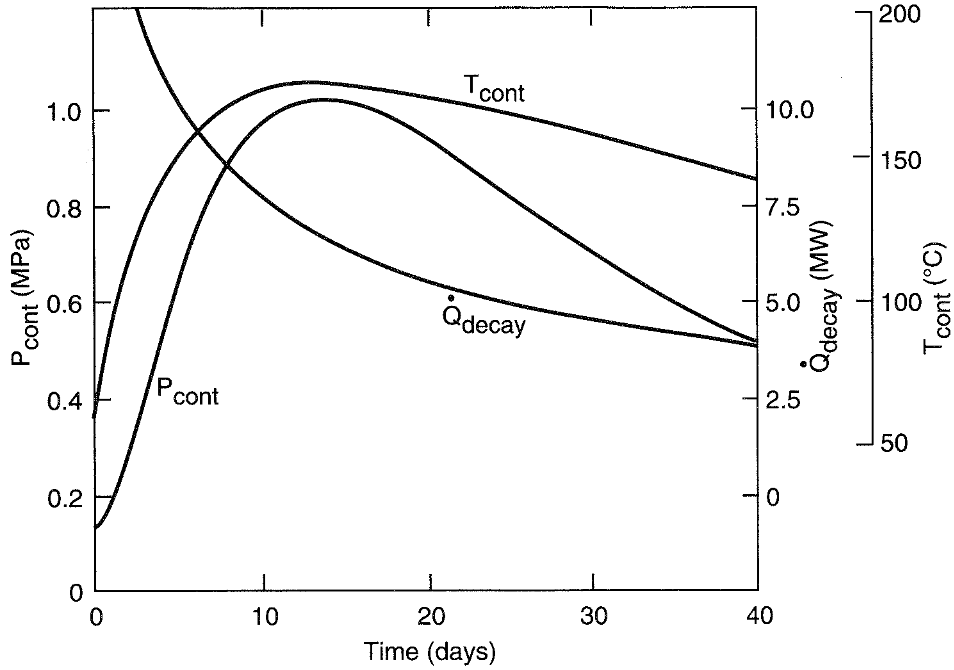


Fig. 16 Filtered containment venting system of a German pressurized-water-reactor plant.



P_{cont} = Containment pressure
T_{cont} = Containment temperature
$\dot{Q}_{decay}$ = Decay heat
Free containment volume = 71.2 m^3
Internal concrete volume = 13.2 m^3
Sump water volume = 1.6 m^3
(Without hydrogen deflagration)

Fig. 17 Long-term pressure and temperature behavior in a 1300-MW(e) pressurized-water-reactor containment after a core-melt accident.

(containment venting not considered). The number of days needed to attain a pressure maximum and what its estimated level will be are questions asked also.

With this background in mind, more in-depth considerations have been made that also include passive mechanisms of decay heat removal from the containment. An obvious solution can be derived from the containment of the CONVOY plants. If the present annulus geometry is expanded by providing special inlet and outlet holes, a structure similar to a chimney can be imagined, as indicated in the composite containment in Fig. 10. This provides for a potential of decay heat removal through natural draft cooling of the containment. If this principle is applied to the LP-path scenario mentioned previously, tentative analyses with the use of the CONTAIN model make visible pressure and temperature plots in the accident atmosphere as represented in Fig. 17.[26] According to those plots the maximum containment loadings (10 bar/180 °C) occur after roughly 12 days; during that time the decay heat power has decreased to about 6 to 8 MW(th).

The relevant computations were made with rather conservative assumptions because of the uncertainties inherent in the heat-transfer models existing for conditions of natural draft. Appropriate experimental projects on passive containment air cooling have begun to restrict the uncertainties and to depart from the conservative assumptions, respectively. Figure 18 shows the layout of the PASCO test facility.[27] The test rig consists of an 8-m-high rectangular channel. One of the channel walls is electrically heated. Here the objectives are to investigate the heat transfer by natural air convection, study methods to enhance the convective and radiative heat transfer, and produce models and heat-transfer correlations for the relatively large dimensions under consideration.

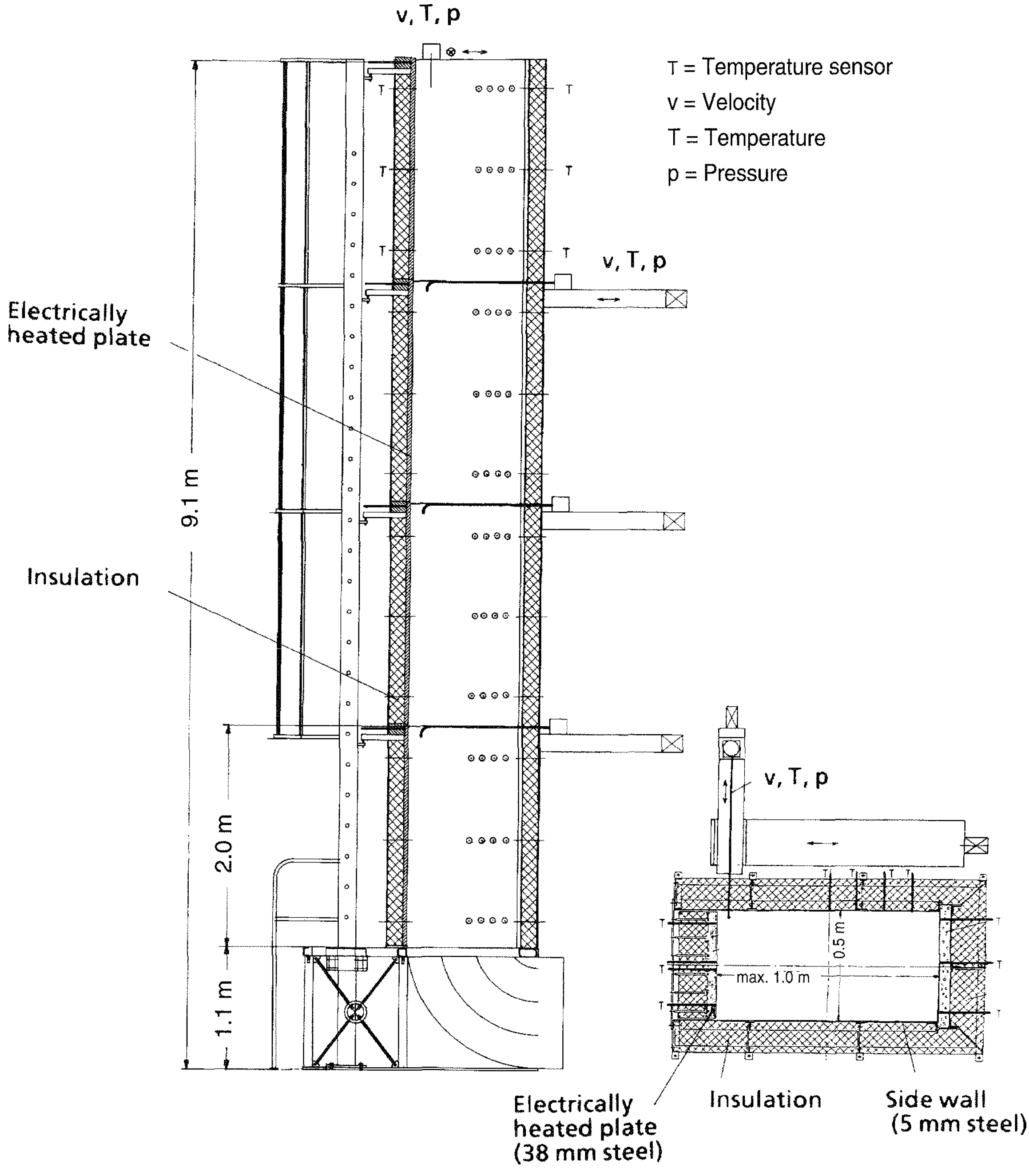


Fig. 18 PASCO test facility for heat-transfer investigations under natural air convection conditions.

Complementing these thermodynamics activities, special development work that relates to a number of variants of vent air filtration in natural draft flow is in progress. With reference to the conditions prevailing in CONVOY plants, a containment leakage rate of 0.25 vol %/day is assumed for reference conditions. In the studies performed so far, no problems have been encountered that would, in a prohibitive manner, cast doubt on the concept outlined previously of a composite containment.[28]

An alternative method of decay heat removal from the containment resembles the system of present-day

CONVOY plants and relies on direct sump-water cooling. The cooling system can be dimensioned in such a way that decay heat removal is feasible largely in a single-phase convection mode. In this way the pressure building up in the containment could be limited to the level of saturation conditions of the sump water. From the engineering point of view, both solutions imply that active and passive means are conceivable. A passive decay heat removal version based on natural convection in the sump-water pool is shown in Fig. 14.

CONCLUSIONS

This article has attempted to present R&D activities and projects that are being carried out at KfK in support of an evolutionary development of the safety features for future PWR plants. These investigations relate mainly to severe accident phenomena, which dominate today's containment integrity challenges.[29] By using a modern PWR as an example, various aspects have been illustrated to strengthen the containment design, with the objective of confining, even in an extremely improbable core meltdown accident, the consequences on the plant.[30] This is in conformity with the general INSAG trend assessment: "Ultimately, (future) plants would be so safe that there would be no technical justification for an emergency plan involving evacuation of the nearby population."[31] This central idea is also a guideline in the German–French R&D cooperation among reactor manufacturers, utilities, and research centers on both sides, with the objective of establishing a joint project of future PWR development.[32,23] In the working groups participating in that cooperative effort, possible solutions of individual problems are being discussed extensively, and proposals might occasionally be evaluated differently. However, these differences are covered by a general agreement regarding the goals to be pursued, namely, to integrate the various preventive and mitigative measures into an innovative balanced overall concept for the safety of future PWR plants so that the provision made against residual risks gains in transparence also for the public.

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Control and Instrumentation

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Upgrading of the K-Reactor Supplementary Safety System at the Savannah River Site

By L. R. Canas, R. L. Garrett, and I. K. Paik[a]

Abstract: *In the event that safety rods fail to scram, the supplementary safety system (SSS) in the K Reactor at the Department of Energy's Savannah River Site nuclear materials production complex provides a second line of defense for attaining shutdown by injecting a neutron poison solution into the moderator space of the reactor tank. Recently the SSS has been upgraded with a secondary poison injection to remedy a potential deficiency of the original design during a drop of coolant flow as a result of a loss of a-c power to the coolant pumps. This article outlines the basis of the functional performance requirements (starting delay, input flow, and flow duration) for the secondary poison injection which ensure the effectiveness of the SSS for any anticipated transients without scram.*

The supplementary safety system (SSS) in the K Reactor at the Department of Energy's (DOE's) Savannah River Site nuclear materials production complex provides a second line of defense for attaining shutdown in the event that safety rods fail to scram. The SSS injects a gadolinium poison solution (GPS), a liquid neutron absorber, which inserts negative reactivity quickly to maintain subcriticality for any credible event that can impair safe operation of the reactor.

Before 1990 the SSS featured GPS injection directly into the moderator space of the reactor tank only and was considered adequate protection for any adverse circumstances. A potential limitation, however, was identified during the analysis of a coolant flow reduction arising from a loss of a-c power to the coolant pumps. This occurrence is a design-basis event included in Chap. 15 of the Safety Analysis Report (SAR) for the K Reactor. In this event the coolant flow decays exponentially over a period of about 2 min from full flow to about 27% of full flow (d-c pump power only). Not only is the GPS dispersion inside the reactor potentially reduced because of the diminishing convective mixing in the moderator space but also the active inventory can be severely reduced because of the longer recycle time in the external coolant loops (because of the decelerating flow) in conjunction with the diminishing rate of GPS injection.

In 1990 the SSS was upgraded with a secondary GPS injection designed to remedy the potential insufficiency of negative reactivity during a loss-of-coolant a-c pump power event. Additional GPS is now injected into the external coolant loops at the pump suction lines. This injection not only provides a shorter path for the GPS to the fuel assemblies of the reactor core (where the material is most effective) but also ensures a plentiful inventory in the reactor at any time.

This article discusses the basis of three key functional performance requirements (FPRs) (maximum starting delay, minimum input flow, and minimum flow duration) for the SSS secondary injection which ensure that the reactor will be safely shut down in the worst scenario of a loss-of-coolant a-c pump power.

SYSTEM DESCRIPTION

K Reactor

A schematic diagram of the K Reactor, the primary coolant system, and the secondary coolant system is shown in Fig. 1.

[a]Westinghouse Savannah River Company, P.O. Box 616, Aiken, SC 29802.

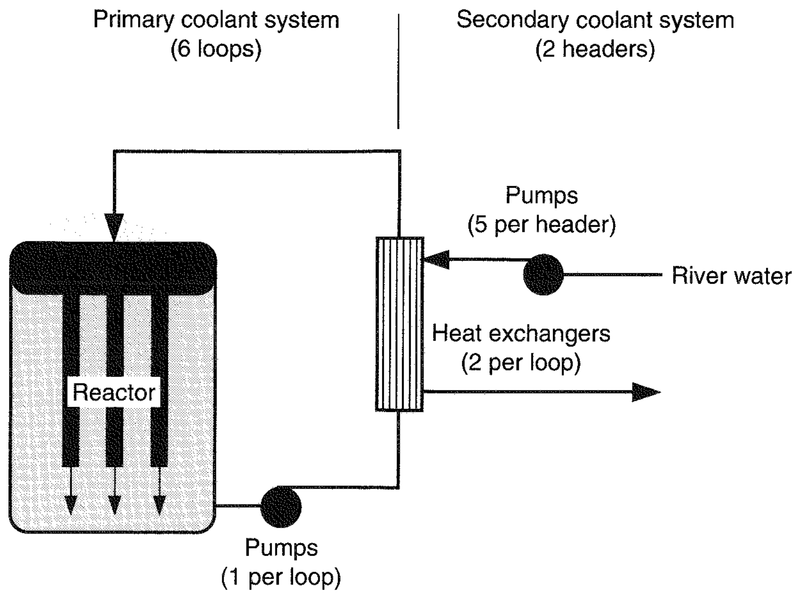

Fig. 1 K Reactor coolant systems.

The primary coolant system has six circulation loops external to the reactor. In each loop approximately 25 000 gpm of heavy water (D_2O) is pumped out an exit nozzle at the bottom of the reactor tank, through two parallel heat exchangers, and into an entrance nozzle to the coolant plenum at the top of the reactor tank. Each pump is driven by an a-c and a d-c motor through a gear reduction box. The d-c motor is powered by a dedicated diesel generator and is intended to provide a backup for the a-c motor and maintain low flow when the reactor is shut down.

The secondary coolant system provides the river water flow that accepts the heat from the hot D_2O flow in the primary coolant system. The two systems are interfaced by 12 heat exchangers (2 for each loop of the primary coolant system). In each heat exchanger the secondary coolant flows in the shell side and the primary coolant in the tube side. River water is pumped from a 25-million-gallon reservoir via two headers, each of which supplies half of the 12 heat exchangers. Each main header carries approximately 85 000 gpm of water.

SSS

The SSS, illustrated in Fig. 2, consists of two redundant trains, each of which features a primary injection (pre-1990 design) of GPS directly into the moderator space of the reactor tank and a secondary injection (post-1990 upgraded design) into the external coolant loops at the pump suction lines. The SSS is designed to shut down the reactor with only one train operable in the event the other one fails.

The primary injection supplies GPS via six symmetrically arranged, perforated tubes called spargers. Each SSS train is associated with three alternating spargers. Each sparger has seven holes evenly spaced along its length up to about the midpoint of the moderator level inside the reactor. As it jets out the sparger holes, the GPS is quickly dispersed throughout the moderator space by the convective flow of coolant out the assemblies. Upstream from the spargers the GPS flows in one of three available paths, depending on the mode of activation. Two of the paths have explosive valves, whereas the remaining path features a pneumatic valve. A particular valve is opened either automatically by the reactor protection systems or manually by operators, depending on the circumstances that demand actuation of the SSS.

The secondary injection feeds GPS to all the six external coolant loops at the pump suction lines. Each SSS train is associated with three alternating loops. The GPS flows in a single path before branching out to the coolant loops. A pneumatic valve in this path is triggered open automatically by the pressure spike of the onset of the primary injection.

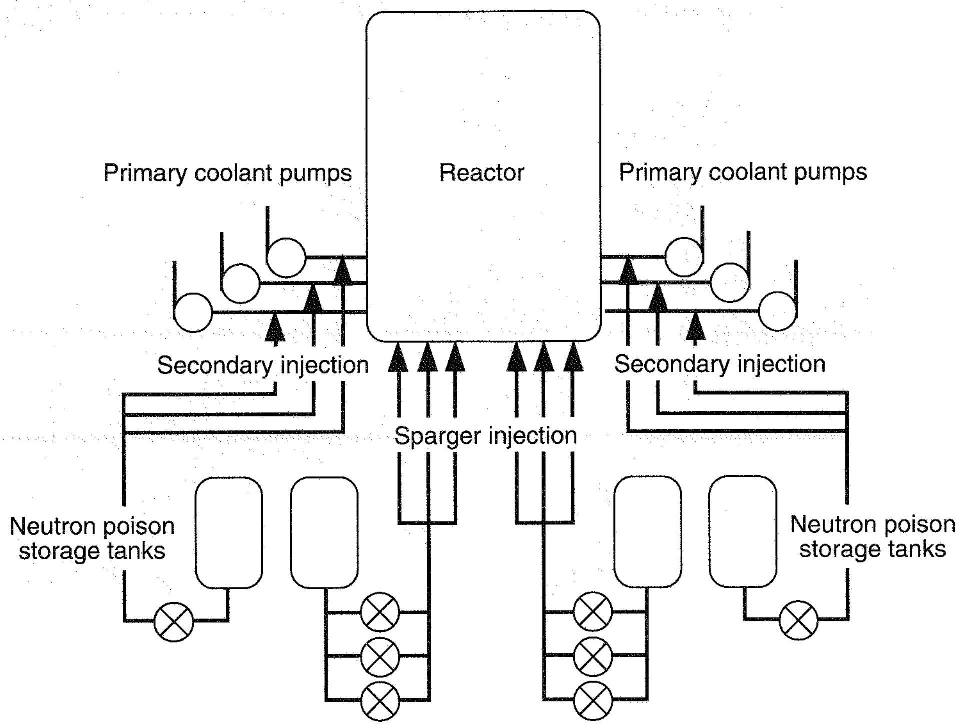

Fig. 2 Supplementary safety system.

The driving force for flow in both the primary and secondary injections is provided by a blanket of pressurized nitrogen over the GPS in the storage tanks.

EVENT DESCRIPTION

Loss-Of-Coolant A-C Pump Power

A loss of a-c power to the K Reactor primary coolant pumps is an event of moderate frequency ($\geq 1 \times 10^{-2}$/yr) and is included in Chap. 15 of the SAR for the K-14.1 charge. Upon occurrence, the coolant flow decays exponentially from full flow (about 25 000 gpm per loop) to about 27% of full flow after about 2 min (Fig. 3). The lower flow is that maintained by the d-c motors alone driving the pumps. The potentially detrimental consequences of this event (caused by the escalating power-to-flow ratio in the reactor core) are normally prevented by the primary protection system, which quickly scrams the reactor by insertion of safety rods. Should the primary protection system fail for any reason, the SSS is relied upon to shut down the reactor by injection of GPS.

For scenarios other than those mentioned previously, for which full coolant flow is maintained, the dispersion of GPS from the SSS primary injection in the moderator space of the reactor tank is quite complete, as promoted

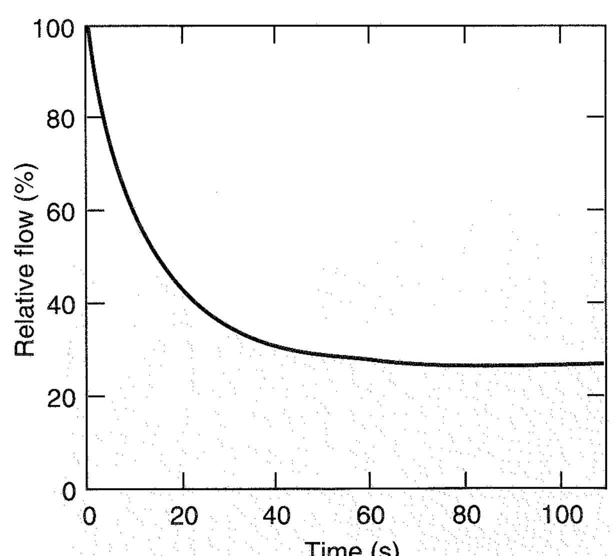

Fig. 3 Primary coolant flow coastdown on loss of a-c pump power.

by the normal coolant-flow profile depicted in Fig. 4. The average residence time of the GPS in the reactor is about 15 seconds, whereas it takes just 12 seconds for exiting GPS to traverse the external loops of the primary coolant system and reenter the moderator space. Hence the primary injection alone can provide a more than adequate inventory of GPS. Before 1990 the SSS consisted of the primary injection only.

In the course of the formal analysis of the loss-of-coolant a-c pump power event for the K-14.1 charge, however, a potential limitation of the primary injection was identified for the aforementioned coasting coolant flow associated with this event. Specifically, the normal coolant flow profile inside the reactor tank flattens as the flow drops and thereby impairs the dispersion of the GPS. This effect is enhanced by the increased coolant density from the quick drop in temperature as the reactor power initially comes down upon injection of the GPS. In the limit, the coolant flow profile in the reactor drops about half way. Hence it is plausible that a transient insufficiency of GPS inventory can develop in the upper half of the reactor. The diminishing GPS inventory can even extend to the entire moderator space as the primary injection becomes exhausted in conjunction with the longer recycle time in the external loops as the coolant flow decelerates. In such a scenario the reactor can potentially regain criticality.

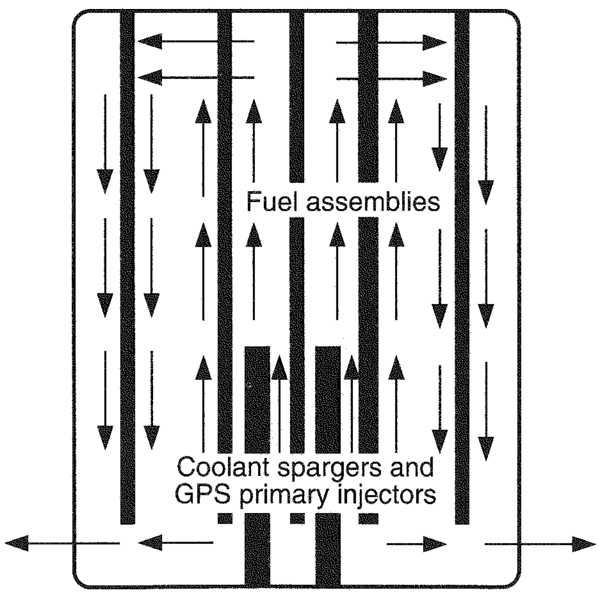

Fig. 4 Coolant flow and gadolinium poison solution dispersion pattern in reactor tank.

SYSTEM MODIFICATION

SSS Design Upgrade

The potential limitation of the primary injection of the SSS in the pre-1990 design prompted Westinghouse (at the request of DOE) to evaluate suitable remedies within a time period that would not severely impact the already tight schedule for the K Reactor restart.

A preliminary analytical assessment of the transient distribution of GPS in the moderator space of the reactor tank during a loss-of-coolant a-c pump power event was performed. In turn, this information was used to simulate the reactor behavior during the indicated event and therefore establish whether or not there was a valid safety concern regarding the performance of the SSS. The results of this effort, however, had a wide band of uncertainty and could not be relied upon for an unequivocal conclusion.

An improvement of the SSS to overcome its potential insufficiency during a loss-of-coolant a-c pump power event was also considered. From technical brainstorming sessions, three potential options resulted: (1) increase the duration of the primary injection, (2) provide a secondary GPS injection into the upper half of the moderator space, and (3) provide a secondary injection into the fuel assemblies via the external coolant loops. The first option was discarded because it did not resolve the potential void of GPS in the upper half of the moderator space. The second option was technically viable but costly to implement in terms of hardware modifications, manpower, and impact on the K Reactor restart schedule. The third option was not only technically viable but also implementable at a relatively low cost within approximately 1 year. Hence the decision was made to enhance the SSS design to provide a secondary GPS injection into the external coolant loops at the pump suction ports.

SSS Functional Performance Requirements

Because the proposed upgrade of the SSS design originated from the preliminary analysis of the loss-of-coolant a-c pump power event for the K-14.1 charge SAR, the Safety Analysis and Engineering Services Group (which had responsibility for all design-basis events in Chap. 15 of the SAR other than for a loss-of-coolant accident) was called upon to prescribe the key FPRs for the secondary injection of GPS to ensure that the safety criteria for the event would not be violated. In short, the SSS must be able to shut down the reactor and maintain a safe margin of subcriticality during the indicated event with a postulated failure of a normal scram by insertion of safety rods.

From the SAR standpoint, the key FPRs for the SSS secondary injection were maximum delay for initiation of the injection, minimum duration of the injection, and minimum input flow of GPS over the duration of the injection. A discussion of the bases for these requirements follows.

Injection Initiation. During a loss-of-coolant a-c pump power event, the primary injection of GPS starts no later than 5 seconds into the event (based on a conservative simulation of reactor behavior with a postulated failure of the primary scram). This injection, in turn, lasts for 30 seconds (conservative cutoff), or equivalently ends at 35 seconds into the event. The secondary injection should therefore start no later than the point in the event time scale that allows the poison front to enter the moderator space of the reactor coincidentally with the tail of the primary injection. This criterion ensures a continued GPS inventory in the moderator space, although potentially limited to the lower half of the reactor, as described previously.

Hence the front of the secondary injection is traced back along the reactor assemblies and external coolant loops to the input ports on the pump suction lines. At the normal coolant flow (about 25 000 gpm per loop), the corresponding transit time has been calculated and experimentally verified at 12 seconds. With the decelerating flow ensuing from a loss-of-coolant a-c pump power, however, this time increases roughly exponentially up to the value at steady d-c flow. If the aforementioned reference point of 35 seconds in the event time scale and the coolant flow decay function in Fig. 3 (conservatively lowered proportionally to its nominal value all along, up to a maximum of 5% of full flow) are used, the required injection start is calculated as no later than 8 seconds in the event time scale. This is shown graphically in Fig. 5.

Injection Duration. Even though the criterion for the maximum delay to the initiation of the SSS secondary injection assures a continued inventory of GPS in the moderator space, the reduced dispersion of poison therein during a loss-of-coolant a-c pump power event (which potentially starves the upper half of the reactor) still precludes absolute insurance against a potential power rebound. This shortcoming is, however, overcome by maintaining the secondary injection long enough to ensure a continued minimum inventory of GPS in the coolant channels of the fuel assemblies.

The criterion for the minimum duration of the secondary injection was therefore conservatively prescribed as the time required for the injection front to complete the travel cycle back to the injection ports. If the injection starts at 8 seconds into a loss-of-coolant a-c pump power event (as established previously), the injection front is

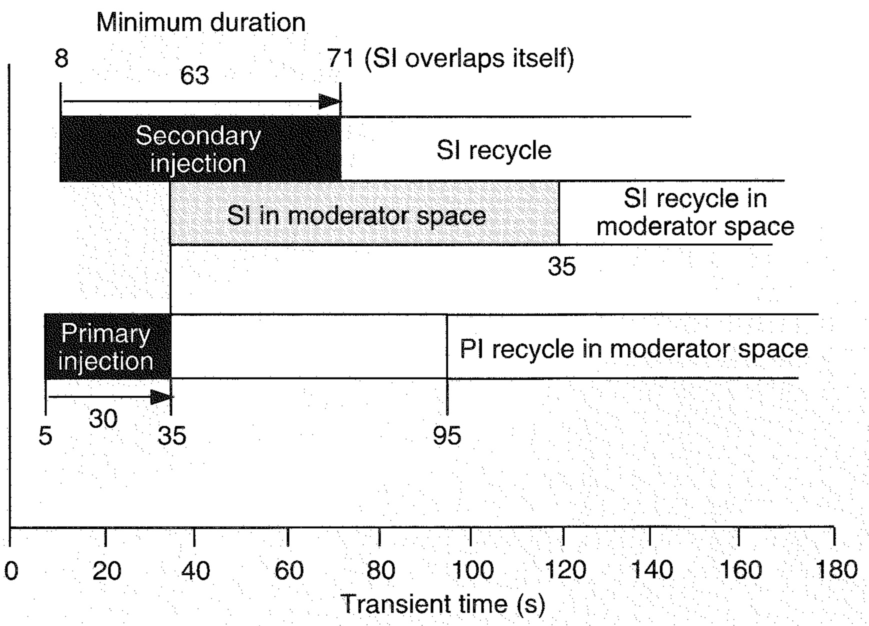

Fig. 5 Supplementary safety system injection event sequences.

followed along an external coolant loop into the reactor plenum, through the fuel assemblies and moderator space, and back into the coolant loop up to the injection port. At the normal coolant flow, the cycle time is about 27 seconds (which suggests an average residence of 15 seconds in the moderator space of the reactor). With the flow coasting down after a loss of a-c pump power, however, the cycle is estimated as 63 seconds (71 seconds in the event time scale following a start at 8 seconds), as shown in Fig. 5.

Injection Flow. As suggested previously, a sufficient inventory of GPS in the coolant channels of the fuel assemblies is still required to ensure that the reactor will stay shut down during a loss-of-coolant a-c pump power event. The criterion for the minimum inventory was selected conservatively to maintain no less than a 1% negative reactivity worth with the following constraints: (1) poison from the secondary injection only is credited, (2) poison in the moderator space (whether from the primary or secondary injections) is not credited, and (3) poison in only three alternate sectors of the reactor is credited. The third restriction is equivalent to assuming that one train of the SSS fails (postulated limiting single failure); hence poison is only injected into three of the six alternate coolant loops. The reference negative reactivity worth was conservatively prescribed to more than counterbalance the positive reactivity arising from the coolant temperature drop when the reactor is initially shut down.

With the aforementioned criterion, the Applied Physics Group calculated the required concentration of poison (in terms of the active component, gadolinium). This value was, in turn, integrated over the coolant volume in the credited fuel assemblies to obtain the required inventory at any time. The minimum input flow for the secondary injection as a function of time (in the event time scale) was then calculated by visualizing a sequence of control volumes passing through the assemblies and tracing them back along the coolant loops to determine the instant they go past the injection ports. Because of the decelerating coolant flow, the transit times for the individual control volumes are not identical. Likewise, the time intervals for the individual segments to cruise past the injection ports are not the same. Both variables increase approximately exponentially with time.

Given the minimum gadolinium concentration (in the form of gadolinium nitrate hexahydrate) required in the poison supply tanks of the SSS, the minimum input flow (in gallons per second) for the secondary injection as a function of time follows readily from the required gadolinium input to the control volumes (same for all), the time points at which they pass the injection ports, and their time intervals to traverse past the injection ports. The resulting profile is shown in Fig. 6. As anticipated, the demand is highest initially and thereafter diminishes exponentially.

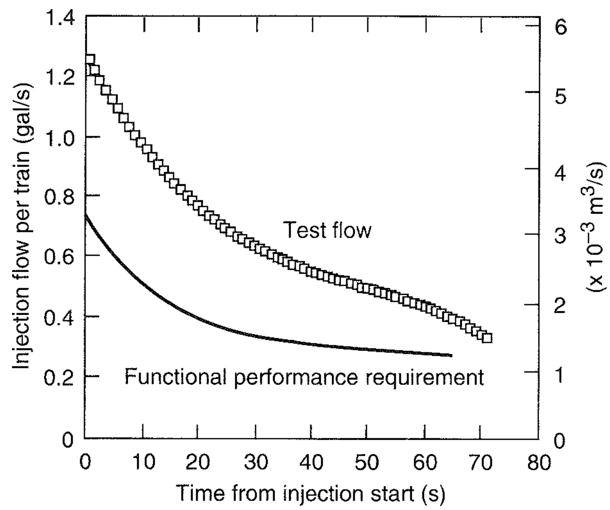

Fig. 6 Supplementary safety system secondary injection flow.

Upgraded SSS Testing

Verification of Functional Requirements. To confirm the derived bases for the stated functional requirements and the reactor response, tests were performed in the K Reactor using the primary and secondary injections of GPS from one train of the SSS. At a power level of 250 MW and normal operating conditions, the SSS was manually initiated by actuation of the explosive valve in the primary injection plumbing. Various parameters, such as assembly effluent temperatures, gamma/neutron flux, and SSS system hydraulics, were recorded as functions of time and compared with predicted responses consistent with the safety analysis.

On the basis of recorded GPS storage tank level data, the time profile of the secondary injection flow was derived and compared with the respective functional performance requirement. The test result in Fig. 6 is evidence that the flow and duration requirements are exceeded.

Validation of Functional Requirements. To confirm that the upgraded SSS fulfills its function, the test further required that a criterion based on fuel assembly coolant temperature be satisfied. The same point kinetics code as that used for simulating many of the design-basis events for Chap. 15 of the K Reactor SAR was used to develop this criterion. Because of the limited neutron detectors available and the difficulties in assessing the spatial neutronics effects of only one SSS train activated, the criterion was solely based on the variable $\Delta T(t)/\Delta T(0)$, where $\Delta T(t)$ is the assembly coolant temperature change as a function of time following the explosive valve actuation and $\Delta T(0)$ is the assembly temperature change at the start of the test immediately before explosive valve actuation. This variable is virtually independent of the initial reactor power and provides a means of direct comparison to the thermal–hydraulic limit that provides the bases of the safety analyses.

As shown in Fig. 7, the typical temperature response for the monitored assemblies was faster than the required temperature drop. This confirms the basis of the derived negative reactivity worth of the SSS applied in the anticipated transient without scram (ATWS) safety analyses.

Supporting Research and Development

GPS Dispersion Study

A study was recently initiated to acquire experimental data on the effects of thermal stratification and reduced coolant flow on the GPS dispersion in the moderator space of the reactor. The data will be obtained in pilot-scale tests at the Idaho National Engineering Laboratories in conjunction with thermal–hydraulic experiments conducted for the benchmarking of the RELAP5 code.

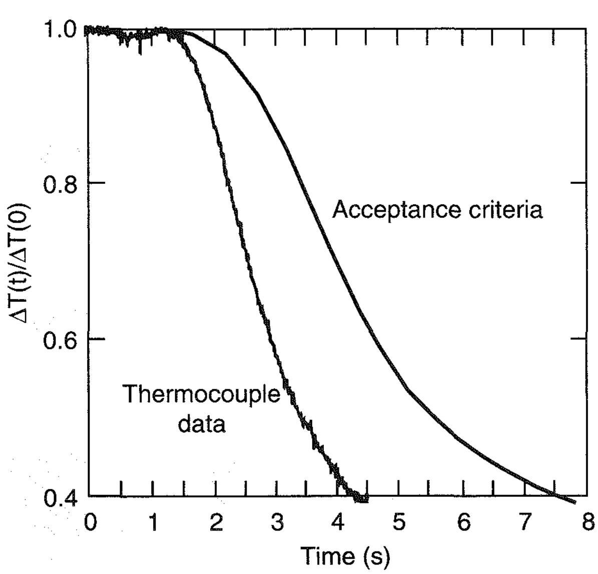

Fig. 7 Assembly delta test.

The experimental facility consists of a pie segment of a reactor tank, a muff, pump suction piping, a pump, and a vertical rise of piping on the pump discharge, all constructed to a one-quarter linear scale and largely transparent for internal visibility and filming purposes. The pie segment is one-sixth of the reactor tank and contains a forest of tubes simulating the fuel assemblies in the actual reactor. The pump is a one-quarter linear scale of an actual coolant pump. Major instrumentation includes water and air flows, water level in the tank at several locations, pressure measurement, photographic flow regime data, and void fractions in the pump suction pipe. This apparatus is presently configured to provide data on and understanding of two-phase flow regimes in a prototypical reactor loop geometry. For the simulation of the GPS injection and subsequent plume dispersion, a scaled sparger is also installed.

Evaluations to integrate the necessary design specifications with the INEL apparatus have been performed. The following series of tests will be performed:

Steady-state verification. Three tests that encompass a scaled full coolant flow with and without sparger operation and a scaled reduced flow with sparger operation will be performed.

Transient verification. Two tests of scaled coolant coastdown will be performed and reproduced for data repeatability.

Temperature transient verification. Two tests with an inlet coolant temperature step change of 25 °C or more and a scaled full coolant flow will be conducted. Dyes of different colors will be used to assess the effects of thermal stratification.

Integrated transient verification. These tests will be performed with a simultaneous coolant-flow coastdown and transient temperature change to assess the effects of flow inertia and thermal stratification. The tests will be repeated for data reproducibility.

CONCLUSIONS

This article illustrates the conception and implementation of a practical and cost-effective solution to a potentially serious problem identified in the course of the safety analysis of a nuclear reactor. The successful resolution of this safety concern with minimum hardware modifications and without impact on the reactor restart schedule translated into cost savings exceeding several million dollars relative to other alternatives. In recognition of this achievement, the authors received the prestigious George Westinghouse Signature Award of Excellence in 1991.

Design Features

Edited by D. B. Trauger

VPBER-600 Conceptual Features and Safety Analysis Results

By F. M. Mitenkov, N. N. Ponomarev-Stepnoi, V. S. Kuul, N. E. Kukharkin, Yu. G. Nikiporets, and O. B. Samoilov[a]

Abstract: *This article describes a Russian approach to designing an enhanced-safety pressurized-water reactor. The basic design philosophy rests on the concept of self-actuating passive systems without the need for active human intervention. An emergency decay heat-removal system independent of normal operating systems and functioning passively is described as well as provisions for ensuring the safety of the system in the event of any accidents resulting in a loss of integrity of the cooling system. The system is designed to have negative reactivity coefficients in all operating regimes and provides passive protection against malfunction by any systems capable of adding reactivity to the reactor. The design uses a guard vessel enclosing the reactor and the primary coolant loop.*

The present stage of nuclear power engineering development, both abroad and particularly in our country, is characterized by the priorities of improving the safety of operating nuclear power plants and creating enhanced-safety reactors for a new generation of nuclear power plants. The outlook for nuclear power development is unambiguously determined by the possibility of ensuring the safety of the population and the environment.

The task of creating a reactor facility with enhanced (in essence, ultimate) safety has been successfully achieved in the form of the AST-500 reactor for nuclear district heating plants. Therefore the principal design solutions that ensure the safety of the AST-500 were taken as a basis for developing the 600-MW(e) VPBER-600 passive safety reactor.

MAIN POSTULATES OF THE VPBER-600 SAFETY CONCEPT

The VPBER-600 safety concept is based on the total implementation of the principles of self-protection for all classes of accidents and on the assurance of safety by means of self-actuating passive systems and devices, which operate without intervention by personnel.

Loss-of-Heat-Removal Accidents

The principal requirements for a totally reliable emergency heat-removal system include the following:

- The system must operate on the basis of natural processes and without the normal power and water supply.
- The system is to be fully autonomous (i.e., independent of normal operation systems as well as of other safety systems).
- The system should not require actuation by human beings (i.e., its actuation is to be accomplished by the use of passive devices on the basis of the direct effects of the reactor parameters).

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• The core is to be cooled by means of natural processes.

• The system must operate for not less than 3 days.

• The primary circuit protection against an inadmissible pressure rise is to be based on the heat-removal principle.

The Primary Circuit Loss-of-Integrity Accident

The principles of safety assurance for loss-of-integrity accidents are formulated as follows:

• Elimination of the classes of accidents involving medium and large leaks, including fracture damage of the reactor vessel.

• Elimination of inadmissible loss of water out of the reactor (keeping the core under water).

• Elimination of the active, fast-acting emergency reactor makeup system and of the emergency core reflood system.

• Localization of radioactive coolant releases with the help of passive localizing systems.

Reactivity Accidents

The pressurized-water-reactor (PWR) self-control and self-protection features are supplemented by the following requirements:

• Assurance of negative reactivity feedback at all operating regimes, including, for these purposes, limiting the concentration of boron in the coolant.

• Elimination of the automatic, simultaneous withdrawal of a large number of the control rods from the core; limitation of the value of the positive reactivity introduced.

• Use of passive protection for systems that affect the activity.

DESIGN SOLUTIONS FOR SAFETY SYSTEMS

The VPBER-600 reactor design is based on a complex of principal solutions that establish a high level of facility protection against loss-of-coolant accidents (LOCAs), which constitute the most hazardous and significant class of PWR accidents.

The reactor facility is located inside a leak-tight concrete containment. The use of an integral arrangement, with the location of all primary circuit equipment inside the reactor vessel, and the use of restrictor devices in the pipelines connected to the reactor make it possible to exclude those classes of accidents which involve large- or medium-size leaks in the primary circuit. The maximum possible size of a failure does not exceed an equivalent diameter of 50 mm and constitutes a small-break LOCA with respect to the characteristics of the transient it would cause.

An additional passive barrier, a guard vessel enclosing the reactor and the primary circuit systems, is introduced to localize the coolant. This vessel is designed to withstand the pressure that develops in the event of a loss of integrity of the primary circuit and thus keeps the core under water, which ensures fuel-element cooling and localization of the radioactivity (Fig. 1).

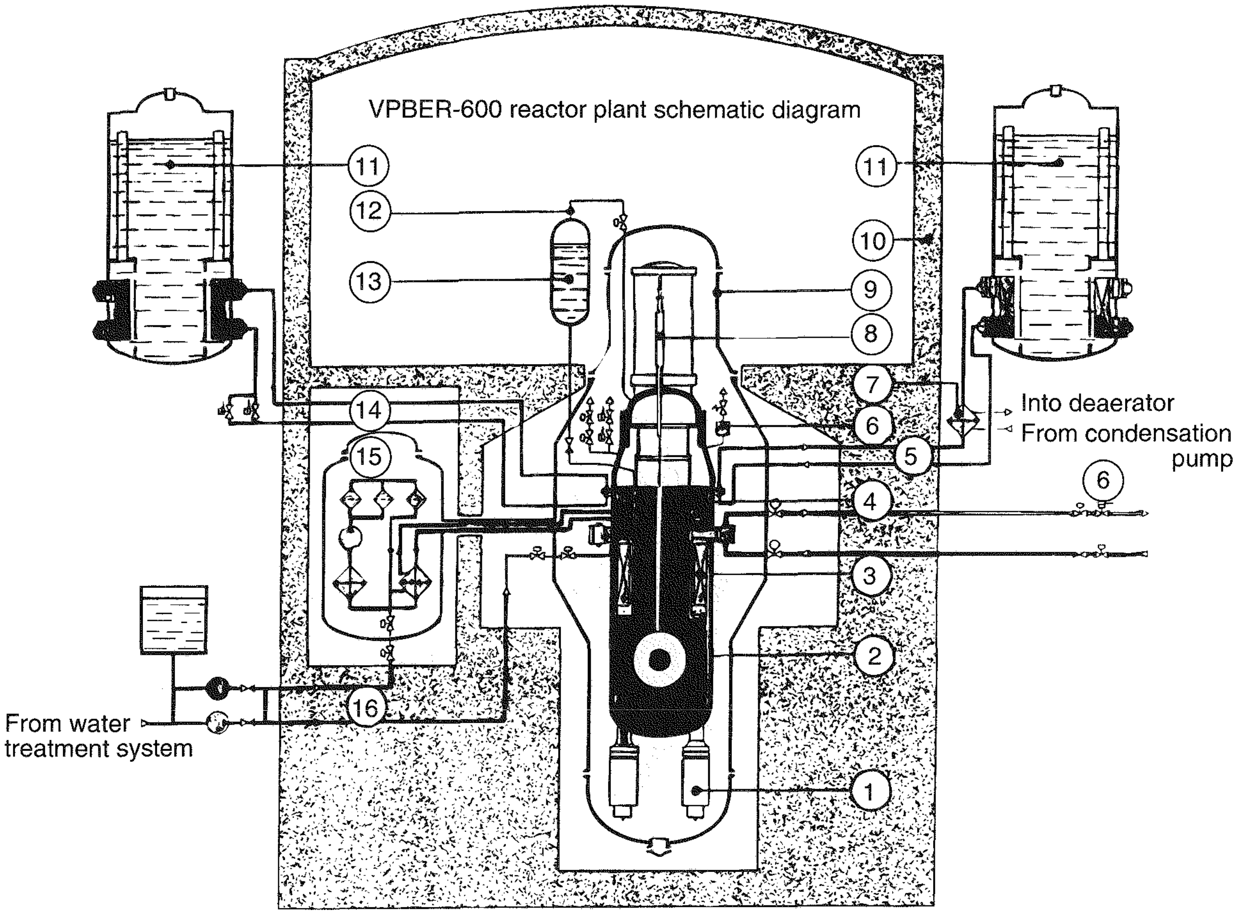
A guard vessel of analogous design, which is used in the AST-500 facility, has gone through extensive tests in the course of design validation and qualification testing. Compartments of marine vessel nuclear facilities that correspond in size and materials to the proposed VPBER-600 guard vessel are serially produced by our domestic industry.

Because of the integral arrangement of the primary circuit, the VPBER-600 reactor coolant circuit is simple in comparison with the traditional loop-type schemes, and thus it produces conditions favorable for natural-convection coolant circulation both under normal coolant-fill conditions and under LOCA conditions.

The continuous heat-removal system (CHRS) and the passive heat-removal system (PHRS), together with the in-vessel heat exchangers, are passive with regard to their operation and actuation. The CHRS operates continuously; the PHRS is actuated by the self-actuated devices. Both are completely independent of the main heat-removal pathway through the steam generators and thus ensure reactor cooling under all classes of accidents, including LOCAs.

The integral arrangement of the VPBER, with the primary coolant inside the reactor, permits a substantial increase in the total heat capacity of the system. The increased capacity allows the core's residual power release to accumulate in the core and thus leads to a long grace period in which accident-management measures may be undertaken. This inertia of the reactor, and consequently the long grace period available, even in case of a severe accident, offers great certainty that the operating personnel will have the capability to manage the accident and thus ensures additional safety.

In addition, self-actuating devices are provided: electric supply breakers on the control and protective system (CPS) drive motors, which, by direct action of the



1. Primary coolant pump
2. Reactor
3. Steam generator
4. Heat exchanger - condenser
5. Continuous heat removal system
6. Self-actuating devices
7. Intermediate heat exchanger
8. CPS drives
9. Guard vessel
10. Containment
11. Heat exchangers unit
12. Liquid absorber injection system
13. Tank with boron solution
14. Passive heat removal system
15. Clean up and reactivity boron compensation system
16. Makeup system

Fig. 1 VPBER-600 reactor plant flow diagram.

pressure in the reactor or in the guard vessel and without the intervention of automatic systems or of personnel actions, ensure that the CPS drives are deenergized and that the reactor is shut down in case of a failure of the electrically controlled emergency protection system. The power supply of the CPS drives is designed for a maximum load that cannot move more than 12 drives at the same time.

The physical properties of the core and the characteristics of the pressurizer make it possible to provide reactor self-shutdown and self-limitation of reactor power even without using the emergency protection owing to strongly negative "power-boiling" coupling.

Coolant-level and coolant-pressure self-actuating devices are also used for (1) the actuation of the PHRS, (2) opening a decompression system that connects the reactor gas space with the guard vessel, and (3) a steam generator localizing system designed to cut off the steam generator if it loses integrity. A passive emergency boron injection system is intended to take the core to a subcritical state and keep it in that state in the event that the CPS drives fail; it also provides for reactor coolant makeup in the event of a LOCA. The system consists of two tanks. Boron solution runs by gravity from one tank, which is located higher than the reactor, and from the second tank by use of the hydraulic accumulation principle. Emergency boron-injection-system pipelines from the reactor to the first valve are enclosed in leak-tight jackets designed for the primary loop pressure.

SAFETY ANALYSIS RESULTS

The results of the safety analysis have shown that, in all design-basis emergencies, deviations of parameters from the normal operating range are prevented by the control system or by actuation of the emergency protection system. Thus the operational safety margins are not exceeded. If there is a loss of integrity of the primary circuit, the core will nevertheless remain covered with water throughout the accident. Neither deterioration of the heat-removal capability nor overheating of the fuel cladding will take place.

In beyond-design-basis accidents, core destruction is prevented by the reactor's inherent safety features (self-protection); the actuation of passive, direct-acting devices; and the functioning of the passive safety systems. The time behavior of the parameters of the facility during a station blackout accident is shown in Fig. 2. The cooldown is ensured without intervention by personnel and without the availability of electric power or water supply for 7 days.

A main circulating pump loss of integrity beyond the design-basis accident was considered, in which the result would be the outflow of coolant from the reactor inlet chamber section into the guard vessel (Fig. 3). The maximum leak size in this event is determined by the main circulating pump shaft seal and is equivalent to an 18-mm-diameter hole. In the event of complete break-away of the main circulating pump nozzle, the effective leak size would be limited to an even smaller value because the pump impeller would obstruct the flow area.

If there is a loss of integrity of the lower plenum of the reactor vessel, the accident is characterized by an even smaller amount of coolant lost from the reactor because the maximum value of the reactor vessel leak rate would correspond to that from a 15-mm-diameter hole.

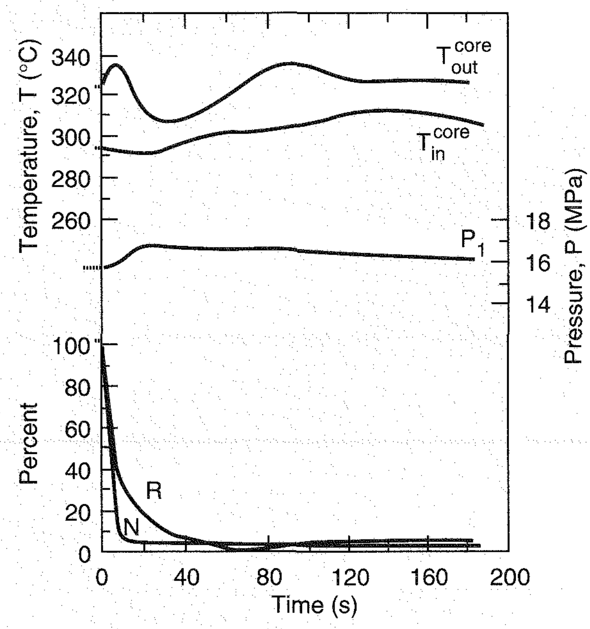


Fig. 2 Plant blackout. T_{out}^{core}, core outlet temperature, T_{in}^{core}, core inlet temperature; P_1, primary cooling system pressure; R, reactor power; and N, neutron flux.

A pressure increase in the guard vessel results in the actuation of the emergency protection system. Self-actuating devices would automatically open the decompression system. Equalization of the pressures in the reactor and in the guard vessel takes place 1.1 hour after the start of the accident. Pressure in the guard vessel reaches a maximum value of 3.2 MPa. Continued cooling of the primary loop results in a gradual pressure decrease in both the reactor and the guard vessel. The core remains covered with the coolant during the entire emergency.

In a beyond-design-basis accident in which a cleanup pipe (maximum diameter pipeline) ruptures and the electric control systems fail, it is postulated that electrical signals would not get through to actuate the emergency protection system, actuate the PHRS, or close the isolating valves on the cleanup system pipeline (Fig. 4). It is assumed, however, that at the same time the normal signals for cutting off the "steam generator on" protective system signal are transmitted. It is also postulated that one PHRS channel fails.

The equalization of pressure in the reactor and the guard vessel stops the water outflow. Restoration of the water inventory in the reactor by draining of the hydroaccumulator ensures that the core is continuously covered with water. The maximum pressure in the guard vessel amounts to 4.7 MPa, which does not exceed the limiting value specified by strength conditions. Thus the leak-tightness of the guard vessel is preserved.

On the basis of the analysis performed, the probability of core damage is less than 10^{-7} per reactor year. The hard-to-imagine conglomeration of system and equipment failures that would be required and the low probability of the events lead to the conclusion that one may consider core damage to be hardly probable from the engineering point of view.

Nevertheless, a series of technical and organizational measures have been adopted in the design to mitigate consequences of this event in the framework of a severe accident management strategy according to up-to-date tendencies and newest design approaches.

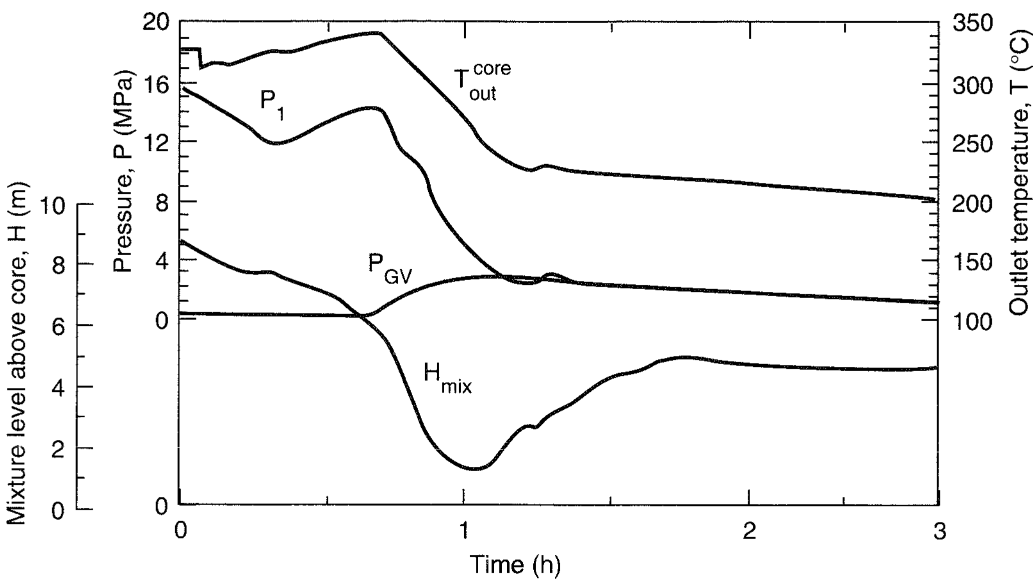


Fig. 3 Loss of integrity in primary coolant pump. T_{out}^{core}, **core outlet temperature;** P_1, **primary cooling system pressure;** P_{GV}, **guard vessel pressure; and** H_{mix}, **mixture level above core.**

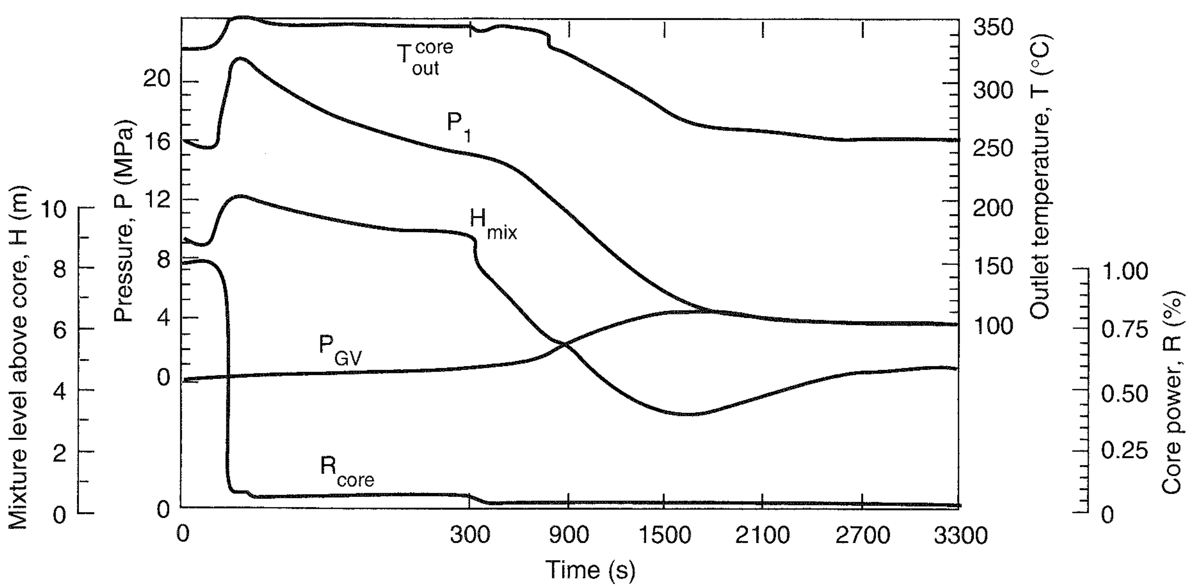


Fig. 4 Cleanup system pipeline rupture with failure of automatic control system. T_{out}^{core}, **core outlet temperature;** P_1, **primary cooling system pressure;** H_{mix}, **mixture level above core;** P_{GV}, **guard vessel pressure; and** R_{core}, **reactor core power.**

Environmental Effects

Edited by E. G. Silver

Radiological Consequence Analyses Under Severe Accident Conditions for the Advanced Neutron Source Reactor at the Oak Ridge National Laboratory[a]

R. P. Taleyarkhan, S. H. Kim, and V. Georgevich[b]

Abstract: *This article discusses salient aspects of methodology, assumptions, and modeling of various features related to radiation exposure and the health consequences from source terms resulting from two conservatively scoped severe accident scenarios. Radiological consequences for a site-suitability scenario based on 10 CFR 100 guidelines also are presented. Consequences arising from severe accidents involving steaming pools and core–concrete interaction (CCI) events combined with several different containment configurations are presented. Results are presented in the form of mean cumulative values for prompt and latent cancer fatality estimates and related cumulative, complementary distribution functions as a function of distance from the reactor site. It is shown that the reactor-site-suitability risk goals are met by a large margin and that overall risk is dominated by early containment failure combined with CCI events.*

The Advanced Neutron Source (ANS) is a user facility in the design stage at Oak Ridge National Laboratory (ORNL).[1] The ANS is planned to be a 330-MW research reactor that uses U_3Si_2–Al cermet fuel in a plate-type configuration. A defense-in-depth philosophy has been adopted. In response to this commitment, the ANS project management initiated severe accident analysis and related technology development early in the design phase to aid in designing sufficiently robust containment for retention and controlled release of radionuclides in the event of an accident. The approach also provides a means for satisfying on- and off-site regulatory requirements, accident-related dose exposures, and containment response and source-term best-estimate analyses for the Level-2 and -3 probabilistic risk analyses that will be produced.

This article presents the methodology, assumptions, and modeling of various features related to radiation exposure and the health consequences from source terms resulting from two conservatively scoped severe accidents. Two containment configurations (viz., early containment failure and intact containment) were considered. Radiological consequences for a site suitability scenario based on 10 CFR 100 guidelines[2] (referred to herein as the CFR 100 scenario) also are presented. Details of the source-term evaluation process and resulting source-term

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characteristics are given in a companion paper (by the authors).[3] The source-term profile for the various cases is summarized in Table 1. Briefly, two different scenarios are analyzed for deriving source terms. The first scenario (i.e., cases SC1-A and SC1-B) evaluated maximum possible steaming loads and associated radionuclide transport for failed and intact containment configurations, respectively, whereas the second scenario (i.e., cases SC2-A and SC2-B) evaluated containment loads and radionuclide transport from a CCI event for failed and intact containment configurations. Preliminary analyses were conducted to evaluate the relative frequency of these events and are shown alongside the various cases in Table 1. As shown, these are very low probability events, the frequencies of which will be taken into account to evaluate the effective risk from these hypothetical severe accidents in the ANS.

MODELING OF OFF-SITE RADIOLOGICAL CONSEQUENCES

This section describes the methodology, assumptions, and modeling of various features related to radiation exposure and the health consequences resulting from source terms for the various accident cases considered.

Modeling Methodology Overview

The source-term profiles outlined in Table 1 were used to evaluate radiological consequences in conjunction with the MELCOR Accident Consequence Code System (MACCS).[4] The MACCS consists of a sequence of mathematical and statistical models that represents radioactive material immediately after release from containment, movement of the material as it disperses downwind of the plant, deposition of the radioactive material onto the ground, and the effects of the airborne and deposited material on humans and the environment. Consequences estimated by MACCS are early health effects, chronic (i.e., latent) health effects, and economic impacts.

Dispersion and deposition of radionuclides released from reactor containment to the atmosphere were modeled with a straight-line Gaussian plume model in MACCS. Plume rise and dry and wet deposition were taken into account. Downwind concentrations of radionuclides up to a distance of 80 km were calculated for each directional sector around the ANS. Radiation doses to on- and off-site populations were calculated with the use of the concentration of radionuclides predicted by the dispersion models. Exposure pathways considered in evaluating early consequences were direct radiation from the passing plume and from radioactive material deposited

Table 1 Released Mass Fractions and Associated Rates of Energy Release

Case	Time, h	Fractional mass release: Xe, Kr	Cs, Na, Rb	I, Br	Te, Se	Energy rate,[a] W	Net occurrence frequency, per year
SC1-A	0 to 4	4.367×10^{-2}	1.340×10^{-6}	1.967×10^{-2}	1.340×10^{-6}	1.823×10^{4}	6.25×10^{-7}
	4 to 12	1.560×10^{-1}	5.212×10^{-3}	1.561×10^{-1}	5.212×10^{-3}	6.555×10^{4}	
	12 to 72	8.290×10^{-2}	1.066×10^{-2}	8.540×10^{-2}	1.066×10^{-2}	4.658×10^{3}	
SC1-B	0 to 10	1.467×10^{-5}	1.652×10^{-9}	4.867×10^{-8}	1.652×10^{-9}	1.181	2.50×10^{-6}
	10 to 72	1.032×10^{-3}	2.334×10^{-7}	6.164×10^{-6}	2.334×10^{-7}	2.526	
CFR 100[b]	0 to 10	1.380×10^{-5}	1.169×10^{-9}	1.461×10^{-8}	1.173×10^{-9}	1.099	2.50×10^{-6}
	10 to 72	1.053×10^{-3}	1.536×10^{-9}	1.677×10^{-6}	1.542×10^{-9}	19.21	
SC2-A	0 to 0.48	8.242×10^{-2}	8.039×10^{-2}	7.768×10^{-2}	8.041×10^{-2}	1.753×10^{5}	3.10×10^{-7}
	0.48 to 1.31	1.316×10^{-2}	1.236×10^{-2}	1.252×10^{-2}	1.235×10^{-2}	1.840×10^{4}	
	1.31 to 2.75	5.320×10^{-3}	4.770×10^{-3}	5.340×10^{-3}	4.750×10^{-3}	5.131×10^{3}	
	2.75 to 20	3.900×10^{-3}	3.080×10^{-3}	3.940×10^{-3}	3.090×10^{-3}	8.438×10^{2}	
SC2-B	0 to 0.48	3.140×10^{-8}	1.50×10^{-10}	1.43×10^{-10}	1.50×10^{-10}	1.032×10^{-2}	1.25×10^{-6}
	0.48 to 1.31	3.842×10^{-7}	6.790×10^{-8}	6.768×10^{-8}	6.795×10^{-8}	1.538×10^{-1}	
	1.31 to 2.75	1.654×10^{-6}	1.344×10^{-6}	1.366×10^{-7}	1.342×10^{-7}	2.317×10^{-1}	
	2.75 to 20	8.419×10^{-5}	3.553×10^{-7}	1.861×10^{-7}	4.355×10^{-6}	3.661	

[a]Power generation from fission products in plume segments.
[b]For the CFR 100 scenario, same release fraction as the Te, Se class is applied for all other nonvolatiles.

on the ground and inhalation of resuspended ground contamination. It is well known that air pathway exposures are dominant contributors to the effects of a severe accident and typically are several orders of magnitude larger than those from liquid pathways. Note that the severe accident scenarios postulated and described in the companion paper (by the authors) already embody a significant measure of conservatism. According to the modeling for these accident scenarios, reactor coolant system (RCS) liquid pathways do not lead to radionuclide transport to the environment and, thereafter, to people. Hence radiological consequences arising from RCS liquid pathways are not modeled explicitly. For the assessment of the long-term impact of water pathways in general (i.e., from rain, rivers, and lakes), values recommended as defaults in MACCS were used with suitable modifications to represent the environment around the ANS.

Emergency response actions were accounted for in the evaluation of potential radiation doses. Short- and long-term actions, such as evacuation, sheltering, and relocation, also were considered, as described subsequently.

MACCS Model for ANS Site and Associated Modeling Assumptions

The proposed ANS site was chosen as the center of a polar grid. The grid was divided into 16 equally spaced sectors (a fixed value built into MACCS) with the outermost radius extending to 80 km. Population data for the various sectors also were developed. Each sector was divided further into 13 elements to reasonably account for the site-specific population distribution. Each element assumes average conditions (i.e., for population, rainfall, wind speed and direction, and radionuclide concentration) in that spatial region. Population data around the ANS site, along with emergency response actions, are summarized in Table 2.

To remain conservative, shielding effects of ANS containment and buildings are not credited. All individuals in the ANS site and within the first four rings receive no shelter unless they have evacuated to the fifth ring, which is the immediate notification zone (INZ). Thereafter the individuals are assumed to be relocated to a safe place and to receive no radiation from the plume. For the first ring (i.e., within the site boundary), it is assumed that all individuals (on the ANS site)

Table 2 Population Distribution and Emergency Response Zones for MACCS Calculations[a]

	Ring and outer boundary designation[b]	Distance, km	Population	Emergency response
1.	ANS site boundary fence	0 to 0.177	449	Evacuation to safety
2.	Exclusion area boundary to 1.6 km (includes HFIR)	0.177 to 1.0	0	Evacuation to safety
3.		1.0 to 1.6	285	Evacuation to safety
4.	LPZ to 2 km	1.6 to 2.0	200	Evacuation to safety
5.	INZ to 3.22 km; ORNL site	2.0 to 3.22	7 006	Sheltering for 6 h and evacuation to safety
6.		3.22 to 4.82	73	Possible relocation
7.		4.82 to 6.44	1 915	Possible relocation
8.	Emergency planning zone to 8.05 km	6.44 to 8.05	15 397	Possible relocation
9.		8.05 to 16.09	70 640	Possible relocation
10.		16.09 to 32.19	241 868	Possible relocation
11.		32.19 to 48.28	288 553	Possible relocation
12.		48.28 to 64.37	140 583	Possible relocation
13.		64.37 to 80.47	144 776	Possible relocation
		Total population	911 745	

[a]Most of the population in this ring are employees located at ORNL. The 2-mile INZ distance also encompasses a handful of residents on private property in Knox County. Emergency actions for these individuals include sheltering in place or relocation to safety.

[b]ANS, Advanced Neutron Source; HFIR, High Flux Isotope Reactor; LPZ, low population zone; INZ, immediate notification zone, and ORNL, Oak Ridge National Laboratory.

will be distributed uniformly over the 16 sectors of the first ring and not consciously positioned in the most unfavorable direction. Although this may sound unconservative, the effect is nullified at least partially via random sampling of the actual weather pattern over the course of one year. In reality, the actual gathering place most likely would be indoors or at a normally upwind outdoor location.

Source terms used for MACCS calculations were derived from the ORIGEN2 and MELCOR evaluations.[5,6] ORIGEN2 calculations for end-of-cycle inventory of radionuclides were used (i.e., for conservatism because fission-product buildup is greatest at end-of-cycle conditions) in conjunction with source-term information for various scenarios. In addition, MELCOR calculations were used to specify the energy content of the generated plumes.

In this model, source terms from various accident scenarios are released at ground level. Such a prescription provides for maximum possible contact with the radioactive cloud before dispersion begins, and, as such, stipulates conservative initial conditions, which may exist for certain accident conditions. Building wake effects are taken into account. Building dimensions are specified to have a width of 66 m and a height of 16 m.

Because of a limitation of MACCS, no credit is given for the ridges and hills surrounding the ANS site, which might block motion of the plume to off-site populated areas. To the extent that ridges and hills can cause greater deposition of aerosol particulates and thus be considered barriers to plume dispersion toward evacuating personnel, the assumption of a flat terrain is conservative. Nevertheless, the effects of the surrounding terrain have been implicitly built into the meteorological data used for dispersion calculations, but the beneficial effects of ridges or hills on dry and wet deposition cannot be accounted for in these data.

Weather data (hourly wind speed, wind direction, and atmospheric stability) taken at the neighboring High Flux Isotope Reactor (HFIR) site tower at a 30-m elevation are assumed to be representative for the ANS site.[7] The ANS is located in the general vicinity of the HFIR with no intervening hills or ridges. The best available data for rainfall and mixing height were used. Rainfall data for the ANS site are assumed to be the same as that for Oak Ridge, and mixing-height data (for morning and afternoon) recommended by the National Climatic Center in Asheville, N.C., are considered representative for the ANS site and surrounding terrain. A weather file consisting of 24 samples per day and 365 days of meteorological information is considered adequate in conjunction with stratified random sampling of four samples per day. [Therefore 1460 (365×4) samples are evaluated for atmospheric dispersion calculations.]

Sixteen kilometers beyond the ANS reactor, boundary weather conditions are applied so that the mixing height is conservatively specified as being at the lowest level (viz., 300 m) from the yearly meteorological data base information supplied by the National Climatic Center in Asheville, N.C. Further, because actual data were unavailable for locations beyond 16 km, we have conservatively assumed neutral stability conditions (i.e., Stability Class D) combined with the specification of no precipitation and a low constant wind speed of 0.5 m/s.

The plume is defined as consisting of multiple sections (i.e., in time) on the basis of guidance received from the source-term transient variation predicted from MELCOR calculations as summarized in Table 1.

On the basis of current emergency procedures for the HFIR site, an evacuation alarm is assumed to sound 10 min after occurrence of a severe accident. Individuals within the first four rings of each sector of the low population zone (LPZ) (i.e., within 2 km) are assumed to start evacuating after a 35-min delay. The 35-min time frame consists of two components. The first component, 30 min, represents the mean time associated with general emergency conditions, including warning employees and visitors to evacuate. This is a standard assumption used previously for similar studies for the New Production Reactor (NPR) Environmental Impact Statement (EIS).[8] The second component, 5 min, represents a reasonable delay between warnings to evacuate and the time people actually start to evacuate. This assumption also is backed up by emergency response drills and actual estimates of the time it would take to send buses to the ANS site for evacuation.

Individuals evacuating from the first four rings of the grid move to safety (i.e., to Ring 5 and beyond) at a speed of 10 m/s (23 mph). Sheltered persons at ORNL are assumed to take 5 min to reach a shelter (after alarm sounds) and then to stay there for 6 h. After this time they receive no more exposure. Upon passage of the plume, sheltered persons may move back to their original spatial element at the end of the emergency phase, which is assumed to last 7 days. This is a standard assumption previously used in similar studies for the NPR EIS.

Relocation of individuals residing outside the INZ is allowed in one of three ways (viz., hot-spot relocation, normal relocation, and long-term relocation). Hot-spot relocation occurs if the effective whole-body dose equivalent to an individual exceeds 0.5 Sv (50 rem) during the 1-week emergency phase. Thereafter individuals in that ring are relocated 30 min after arrival of the first plume. Relocated individuals receive no further dosage during the emergency phase. Normal relocation is activated if the effective whole-body dose equivalent exceeds 0.25 Sv (25 rem) in the 1-week emergency phase. Thereafter individuals in that ring are relocated 1 h after arrival of the first plume at that distance. Relocated individuals receive no further dosage during the emergency phase. Long-term relocation is activated if exposure exceeds 0.01 Sv/year (1 rem/year). These assumptions are based on guidance given from default values suggested in MACCS (which also were used for the well-known NUREG-1150 studies).

The breathing rate of individuals is conservatively assumed to be constant and equal to the MACCS default value of 2.66×10^{-4} m^3/s, which is an averaged value near the upper limit of 3.1×10^{-4} m^3/s, as suggested by Nuclear Regulatory Commission regulatory guides.

Other parameters that enter the calculational process, such as protection factors for inhalation or skin exposure, resuspension, cloud and other shielding factors, and specific input required for deriving chronic (i.e., latent) effects, are assumed to be the default values recommended in the *MACCS User's Guide.*

Because of a limitation in MACCS, all the plume transport characteristics for gases, vapors, and aerosols are required to be represented as aerosols For the simulation of the dynamics of the various species, several modeling assumptions had to be made. Noble gases–related aerosols were constrained to be unamenable to wet or dry deposition. The bin size was chosen to be extremely small. Hence noble gases are treated as extremely small aerosols that do not undergo dry or wet deposition and, as such, always remain suspended. Halogen-class aerosols are modeled as being amenable to wet deposition but not to dry deposition. This simulates vapor transport processes. The remainder of the classes are treated as conventional aerosols, which are amenable to both wet and dry deposition.

For modeling off-site consequence calculations for the CFR 100 case, no evacuation or relocation is allowed. Health consequences corresponding to the 95th percentile will be reported as prescribed by 10 CFR 100 guidelines.

RADIOLOGICAL CONSEQUENCE RESULTS AND ANALYSES

Presented in this section are the radiological consequences based on the MACCS model developed for the ANS for source terms outlined in Table 1. Tables 3 to 5 summarize key results of mean-value estimates for health consequences and risks for the various cases as a function of distance. Results also were generated conventionally as Complementary Cumulative Distribution Functions (CCDFs). Stated simply, CCDFs show variations between an event, "X," and one minus the probability of this event, "Pr," occurring (i.e., Pr > X). Selected CCDFs are shown in Fig. 1. For each scenario, CCDF plots were generated for displaying probability variations (i.e., Pr > X) for different events (X) over different distance intervals. These should be used in conjunction with the health consequence results reported in Tables 3 to 5. Note that, for all cases (except the CFR 100 scenario), Tables 3 to 5 show mean values for various consequence parameters. For the CFR 100 scenario, 95th percentile values are listed. The CCDF plots should be used to note important variational trends from mean-value estimates for each of the three distance zones.

Table 3 shows mean cumulative values for prompt and latent cancer fatality estimates as a function of distance from the ANS site. As shown, prompt fatality values are a small fraction of the total number of individuals on site, even for the CCI cases with containment failure (i.e., for Scenario SC2-A), if 449 individuals are assumed to be within the ANS site boundary (i.e., within a radius of 170 m). This can be attributed to the weather patterns at the ANS site and the improbability that all 449 individuals would be in the direct pathway of the plume. As noted, the CCI case provides for greater fatalities than the steaming pool case (i.e., Scenario 1). Indeed, for the CFR 100 case and all Scenario 1 cases, no prompt fatalities are predicted because, for Scenario 1, several hours elapse before any significant amounts of radioactivity are released to the environment, which leaves sufficient time for evacuation and sheltering of all individuals on the ANS site and within the neighboring three rings.

Cancer deaths and injuries are also much smaller for the CFR 100 scenario and Scenario 1 cases compared with those for the Scenario 2 cases. In general, for the steaming pool cases, this is attributed to the time span available for safe evacuation, as mentioned previously, in conjunction with prompt fatality estimates. For the CFR 100 scenario in particular, the low values of health consequences essentially are a result of the leak-tight

Table 3 Mean Values for Health Consequences for Various Accident Scenarios

Distance, km	Health consequences	Scenario: SC1-A	SC1-B	SC1-C	CFR 100	SC2-A	SC2-B	SC2-C	SC2-AF	SC2-BF	SC2-CF
0 to 1.0	Prompt fatalities	0	0	0	0	7.80	0	4.66×10^{-3}	1.79×10^{-1}	0	0
	Cancer fatalities	8.25×10^{-1}	4.49×10^{-5}	1.57×10^{-1}	5.29×10^{-5}	17.1	8.53×10^{-5}	4.13	10.4	1.63×10^{-4}	1.07
	Cancer injuries[a]	5.22	1.53×10^{-4}	5.66×10^{-1}	1.83×10^{-4}	98.4	2.36×10^{-4}	25	60.6	4.36×10^{-4}	5.91
0 to 2.0	Prompt fatalities	0	0	0	0	7.80	0	4.66×10^{-3}	1.79×10^{-1}	0	0
	Cancer fatalities	9.19×10^{-1}	5.36×10^{-5}	1.71×10^{-1}	6.34×10^{-5}	18	9.28×10^{-5}	4.28	10.9	1.82×10^{-4}	1.23
	Cancer injuries	5.53	2.15×10^{-4}	6.08×10^{-1}	2.80×10^{-4}	103	2.57×10^{-4}	25.8	62.7	5.18×10^{-4}	6.39
0 to 3.2	Prompt fatalities	0	0	0	0	7.80	0	4.66×10^{-3}	1.79×10^{-1}	0	0
	Cancer fatalities	1.46	6.92×10^{-5}	2.32×10^{-1}	8.43×10^{-5}	19.5	1.12×10^{-4}	4.59	12.8	2.12×10^{-4}	1.63
	Cancer injuries	8.99	3.01×10^{-4}	9.50×10^{-1}	4.06×10^{-4}	112	3.33×10^{-4}	27.5	71.7	6.35×10^{-4}	7.76
0 to 8.0	Prompt fatalities	0	0	0	0	7.80	0	4.66×10^{-3}	1.79×10^{-1}	0	0
	Cancer fatalities	2.50	1.58×10^{-4}	3.54×10^{-1}	1.16×10^{-4}	20.7	1.53×10^{-4}	4.92	15.3	3.10×10^{-4}	2.35
	Cancer injuries	15.8	7.89×10^{-4}	1.70	8.63×10^{-4}	117	4.83×10^{-4}	28.7	82.7	1.09×10^{-3}	10.3
0 to 80.0	Prompt fatalities	0	0	0	0	7.80	0	4.66×10^{-3}	1.79×10^{-1}	0	0
	Cancer fatalities	19.1	1.06×10^{-3}	1.91	1.86×10^{-4}	42.4	5.54×10^{-4}	8.50	52.3	1.26×10^{-3}	9.38
	Cancer injuries	15.3	5.33×10^{-3}	1.40×10^{1}	1.59×10^{-3}	234	2.45×10^{-3}	46.9	293	5.38×10^{-3}	46.8

[a]Cancer injuries imply cancer of the stomach, lung, thyroid, and skin. For CFR 100 case, 95th percentile values are used.

Table 4 Variation of Average Individual Risk for Steaming Pool Cases[a]

Distance, km	MACCS cancer fatality risk estimates			Effective cancer fatality risk estimates		
	CFR 100	SC1-A	SC1-B	CFR 100	SC1-A	SC1-B
0 to 0.2	9.9×10^{-8}	1.73×10^{-3}	8.37×10^{-8}	2.49×10^{-13}	4.33×10^{-9}	2.09×10^{-13}
0.2 to 1.0	1.1×10^{-8}	2.40×10^{-4}	8.27×10^{-9}	2.80×10^{-14}	6.0×10^{-10}	2.07×10^{-14}
1.0 to 1.6	3.9×10^{-9}	8.78×10^{-5}	3.02×10^{-9}	9.88×10^{-15}	2.21×10^{-10}	7.55×10^{-15}
1.6 to 2.0	2.5×10^{-9}	5.62×10^{-5}	1.98×10^{-9}	6.28×10^{-15}	1.41×10^{-10}	4.95×10^{-15}
2.0 to 3.2	1.4×10^{-9}	9.97×10^{-5}	1.59×10^{-9}	3.60×10^{-15}	2.49×10^{-10}	3.98×10^{-15}
6.4 to 8.0	3×10^{-10}	8.07×10^{-5}	5.24×10^{-9}	8.23×10^{-16}	2.02×10^{-10}	1.31×10^{-14}
64.4 to 80.5	1.04×10^{-13}	2.64×10^{-6}	8.85×10^{-11}	2.60×10^{-19}	6.60×10^{-12}	2.21×10^{-16}

[a]Estimates of MACCS fatality risk estimates assume a probability of 1 for occurrence. Estimates of effective fatality risk estimates include probabilities of occurrence (Table 1). Risk estimates are mean values (fatalities/year).

Table 5 Variation of Average Individual Risk for CCI Case[a]

Distance, km	MACCS fatality risk estimates			Effective fatality risk estimates		
	SC2-A case		SC2-B case	SC2-A case		SC2-B case
	Prompt	Cancer	Cancer	Prompt	Cancer	Cancer
0 to 0.2	1.74×10^{-2}	3.81×10^{-2}	1.79×10^{-7}	5.22×10^{-9}	1.14×10^{-8}	2.24×10^{-13}
0.2 to 1.0	1.05×10^{-5}	3.59×10^{-3}	1.49×10^{-8}	3.15×10^{-12}	1.08×10^{-9}	1.86×10^{-14}
1.0 to 1.6	0	9.57×10^{-4}	4.98×10^{-9}	0	2.87×10^{-10}	6.23×10^{-15}
1.6 to 2.0	0	4.99×10^{-4}	3.17×10^{-9}	0	1.50×10^{-10}	3.96×10^{-15}
2.0 to 3.2	0	3.46×10^{-4}	3.51×10^{-9}	0	1.04×10^{-10}	4.39×10^{-15}
6.4 to 8.0	0	8.75×10^{-5}	2.75×10^{-9}	0	2.63×10^{-11}	3.44×10^{-15}
64.4 to 80.5	0	3.04×10^{-6}	5.73×10^{-11}	0	9.12×10^{-13}	7.16×10^{-17}

[a]Estimates of MACCS fatality risk estimates assume a probability of 1 for occurrence. Estimates of effective fatality risk estimates include probabilities of occurrence (Table 1). Risk estimates are mean values (fatalities/year).

nature of ANS containment, which leads to a relatively insignificant source term (and that, too, over a very long time). Overall, cancer fatalities and injuries between the CCI cases also display the same trend for individuals within the site boundary (i.e., <1 km). Finally, upon comparing cancer deaths and injuries caused by containment failure with those occurring when the containment stays intact, a general spread of 4 to 5 orders of magnitude exists. This underscores the importance of maintaining containment integrity.

An evaluation also was made for the total number of individuals exceeding various levels of radiation doses, as well as a breakdown of total and individual doses to various body organs (bone marrow, lungs, and whole body), for each of the cases. For the site-suitability-basis scenario (CFR 100), the permissible limits are not exceeded for the three body organs. Essentially, this is because of the leak-tight nature of the dual containment of the ANS. The same also was true for all cases where containment isolates and functions as designed. As may be expected, for cases where both primary and secondary containment have failed, the mean number of individuals exceeding the 0.05-Sv (5-rem) Protective Action Guidelines (PAG) limit ranged in the several thousands. For the lung and bone marrow limits (viz., 1.5 and 5.0 Sv), however, only the SC2-A case was found to be significant. For all cases, the individual dose was found to decrease rapidly away from the ANS site. For the steaming pool cases, only the cases where containment failure has occurred were significant. The 0.25-Sv (25-rem) and 0.05-Sv (5-rem) PAG limits for the thyroid and whole-body dose limits are exceeded only for the SC1-A case, where the primary and secondary containments have failed.

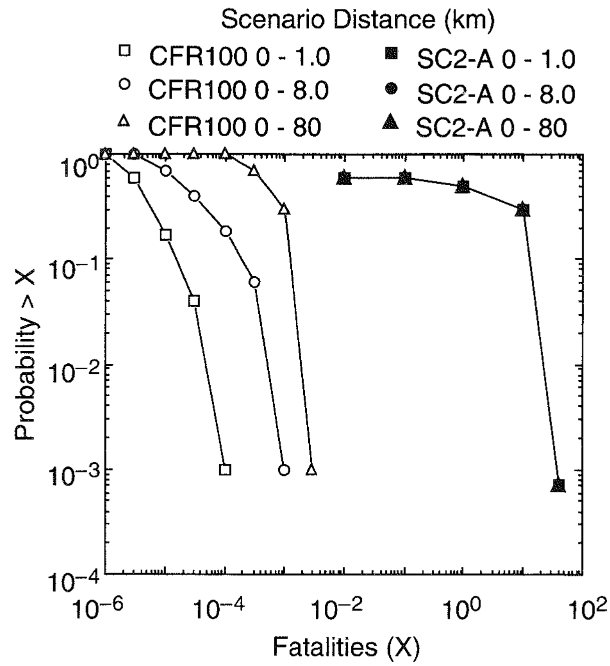

Fig. 1 Prompt (for SC2-A) and cancer (for CFR 100) fatalities, X, CCDFs for CFR 100 and SC2-A scenarios.

Tables 3 and 4 contain a summary of average individual risks (prompt and latent) from the two scenarios, assuming a 100% frequency of occurrence (i.e., MACCS-evaluated risk values for Scenarios 1 and 2) and accounting for the frequency of occurrence of the two scenarios (i.e., effective fatality risk) for various rings in the polar grid. As expected, Scenario 2 cases dominated the risk of prompt and latent cancer fatalities. Note that the MACCS-calculated risk for prompt and latent cancer fatality values shown in these tables are not measures of actual risk. As mentioned in the introduction, to obtain estimates of effective risk, fatality risk estimates presented for the accident scenarios representing early containment failure should be multiplied by the conditional probability (i.e., the net occurrence frequency) for each case as tabulated in Table 1. The columns in Table 5 under the heading "Effective Fatality Risk" reflect this aspect.

Most likely, a further reduction in source terms will occur from removal of conservatisms via best-estimate evaluations (which then would lead to lowering of fatality estimates). On the basis of the results shown in Tables 3 to 5 and Fig. 1, the ANS risk goals in individual categories shown in Table 4 are met with a very wide margin for all cases analyzed under the various assumptions mentioned previously. For overall risk, several additional severe accidents in various release categories must be considered. However, the risk from the other accidents is expected to be lower than the risks highlighted in Tables 3 and 4 and in the CCDFs.

CONCLUSIONS

This article has presented the methodology, assumptions, and modeling of various features related to radiation exposure and the health consequences from source terms resulting from two conservatively scoped severe accidents. This was done for scenarios with two containment configurations (viz., early containment failure and intact containment). For the site-suitability transient case (CFR 100), the radiological consequences and risks are negligible and well within regulatory guidelines and ANS risk goals. Risk and health consequences are dominated by the CCI case coupled with early containment failure. Prompt fatalities were calculated only for the CCI case with early containment failure but not for any of the steaming pool cases. Prompt fatalities are a small fraction of all on-site workers. These features are a result of the weather patterns around the ANS site coupled with the time available for evacuation. Risk dominance for the CCI events is caused principally by insufficient time for evacuating on-site workers. For all cases, ANS risk goals were met by a wide margin.[9]

Actions are being taken to minimize the fatality risk resulting from severe accidents involving early containment failure. Efforts under way include (1) introduction of mitigative features to prevent CCI occurrence and introduction of missile shields to prevent containment failure and/or to allow sufficient time for evacuation and (2) a best-estimate evaluation of core-melt progression. When these are accounted for, the ANS reactor system is expected to be safe, both from probabilistic and deterministic standpoints (i.e., negligibly low values of risk and no fatalities or injuries if a severe accident occurs).

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Waste and Spent Fuel Management

Edited by E. G. Silver

Activities Related to Waste and Spent Fuel Management

Compiled by M. D. Muhlheim and E. G. Silver[a]

This feature includes brief reports on administrative, regulatory, and technical activities related to research, development for, and implementation of facilities and technologies related to the safety aspects of the management of radioactive wastes and spent nuclear fuel.

The information in this issue of *Nuclear Safety* was received during October, November, and December of 1992.

ACNW REPORTS ON SEVERAL ISSUES

During the period covered in this issue of *Nuclear Safety*, the Nuclear Regulatory Commission (NRC) received a number of reports from its Advisory Committee on Nuclear Waste (ACNW). Several of the reports are included.

Significant Issues in the High-Level Waste Repository Program

The ACNW was asked to identify significant issues that have the potential for delaying or otherwise interfering with the timely development of a repository for high-level nuclear waste (HLW). The ACNW focused on items of large scope that could hinder the development of an HLW repository, severely impact the schedule set by the Department of Energy (DOE), or disrupt the orderly licensing process by extensive delays or untimely polemics. In addition, the ACNW was asked to provide an outline of the process of developing an HLW repository. Their report in response to this request reads, in part:[1]

> The issues that appear to qualify for inclusion in this communication constitute a fluid assembly because various parties to the HLW repository program are engaged in ongoing analytical studies, research, development, demonstration, full-scale tests and the like. Further, many studies and other activities are not clearly visible or the outcome of these efforts is not predictable. Therefore, we provide this communication with the caveat that the issues believed to be important today may not be so in the near future. In addition, the Committee provides a summary in which the issues cited in this communication are ordered by the Committee according to their impact on the outcome of the repository development process. Finally, the impact of the recently passed legislation under the Energy Policy Act of 1992 is likely to result in further uncertainties about the relevance of some of the issues raised in this communication.

1. A number of issues have been identified under the heading of regulatory considerations pertinent to site characterization and licensing of a repository.
   a. The NRC staff should develop positions that can serve as a basis for recommendations to the National Academy of Sciences relative to the Academy's role, mandated by the Energy Policy Act of 1992, of providing findings and recommendations on reasonable standards for the protection of public health and safety for the proposed HLW repository at Yucca Mountain.
   b. It is likely that regulations issued by the NRC and other agencies will not be wholly compatible or consistent. It is not clear what constitutes resolution of the issue of compatibility and the stage at which this should be accomplished. The Commission should

[a]Oak Ridge National Laboratory.

request the NRC staff to clarify this issue and, if appropriate, initiate rulemaking. . . .

d. Considerable data that are useful or necessary for a licensing application and are anticipated to be involved in the licensing process will be or have been obtained without use of the rigorous quality assurance procedures now being implemented. The Licensing Support System has been established to encompass pertinent data but has not yet been inaugurated. Further, the LSS may contain data or results that have similar deficiencies. Also, the guidance for the application of QA procedures to development and validation of models, and to decision-making among competing conclusions is at present substantially absent. The inclusion of QA deficient data or protocols in selection, validation and evaluation of uncertainties in models could pose significant difficulties in the licensing process. The Commission should request the NRC staff to initiate a comprehensive review of the guidance to the DOE that is necessary to define the quality requirements for the use of all important data obtained prior to promulgation of the QA requirements and for relevant models developed for the licensing-related repository description.

e. Expert judgment will be a necessary and important part of the licensing process. Acceptance of expert judgment, its methodologies and its results in the waste management arena continues to be controversial and could disrupt a licensing process. The Commission should request the NRC staff to proceed with rulemaking to delineate the processes and standards for application of expert judgment to ensure that this technique can make a useful contribution to the licensing process and that its application will be accepted in an adversarial setting.

f. The NRC staff has apparently taken the position that performance enhancement of the engineered barrier system cannot be used to offset the potential deficiencies likely to be encountered in the geologic media. This position has caused significant concept and design difficulties, appears to be without technical justification and also appears to be without bases in regulations. Owing to the inability to predict for any site if all of the attributes will meet all regulatory requirements, the Commission may wish to examine this position to ensure that the DOE is not burdened with a requirement that is neither necessary nor feasible to implement, and with one that contributes little additional assurance of protection of the health and safety of the public. The Commission should instruct the staff to devise means to ensure that major improvements in the EBS can and should be used to offset inadequate retention/confinement properties of the geologic environment of the waste. The NRC staff should identify functional criteria for such trade-offs.

g. The properties of HLW that was previously stored in pools or dry storage and is assumed to constitute a waste form suitable for disposal in a repository are uncertain. The Commission may wish to require the NRC staff to identify those properties of the stored spent fuel that are of importance to the repository and those tests that are considered necessary for qualification of this waste as the interim storage time lengthens. Similar considerations should also be given to HLW glass that may have been stored for some time under various conditions.

h. A significant part of the licensing process for an HLW repository involves the selection and analysis of scenarios of postulated events in the repository, coupled with the application of a variety of models of the physical system. The processes by which models are designed, tested and, where appropriate, validated to be representative of the present and future behavior of parts of the repository system are not included in regulations or guidance to DOE. Particularly, the protocols for obtaining agreement that a specific model adequately describes the future state of a system have not been defined. The Commission should request the staff to define a methodology for obtaining agreement on this issue in advance of the licensing process. We recommend that this topic be included in early rulemaking, in order to provide guidance to DOE for the performance assessment process.

i. The Environmental Protection Agency regulations have not been codified, and considerable uncertainty remains about the existing standards for ^{14}C and other gaseous radionuclides. In addition, the NRC has not developed specific and comprehensive guidance to DOE on its requirements for the confinement of such radioactive material. This uncertainty could strongly influence the entire EBS design, testing and analysis. The Commission may wish to instruct the NRC staff to begin development of such guidance in the near future, recognizing that the new environmental standards will influence the details of such guidance.

j. Protocols for testing of the EBS and its components under repository-relevant conditions have been difficult to define and apparently such testing has not been conducted in a manner agreed to be satisfactory. The DOE, as well as the Center for Nuclear Waste Regulatory Analyses, has initiated tests that are believed to be repository-relevant. Owing to the extensive time requirements for tests whose results are to be extrapolated over the expected life of the EBS, the Commission should initiate development of guidance, perhaps in the form of staff technical positions, on the criteria for determining when test conditions are repository-relevant.

k. The DOE has indicated that the overall performance assessment of the repository system may not include an allocation from the performance of the waste form. This approach apparently does not agree with the view of the NRC staff and has resulted in exchanges that appear to be at an impasse. Since the waste form (spent fuel, glass) is now either prepared or in the process of being prepared in facilities that are substantially completed, the Commission should request the NRC staff to clarify the details of this disagreement and adjudicate, at an early stage, the position it wishes to take in this matter.

2. The Monitored Retrievable Storage Facility has received attention by the Congress, DOE, various Indian Tribes, cities, counties, and States, but has not developed into an accepted project with a currently valid starting point or a schedule for its completion, licensing and operation. Owing to the pivotal position of the MRS in the disposal of spent fuel, several issues are pertinent.

   a. The required life of the MRS needs to be defined and the specifications, criteria for siting and construction, the content of licensing documents, and the anticipated licensing process need to be established, published and approved. The Commission should request the NRC staff to develop the details of regulations related to the licensing of an MRS.

   b. There has been no substantial development of a backup concept to the MRS in the event that it is not feasible to locate, site, license, or operate such a facility. While the reasons for such a failure will be nontechnical, their effect could be profound. There has been little planning for this eventuality, and the Commission should request the NRC staff to initiate such studies in cooperation with the DOE and the Office of the Nuclear Waste Negotiator.

3. The scientific/technical investigations for the repository program being conducted by DOE are aimed at a comprehensive licensing document for NRC review. The studies that have been completed and those that are in progress are likely to produce results of variable quality or applicability. Further, there will certainly not be enough time and resources devoted to these studies to provide full insight into all scientific/technical questions. The NRC staff has commented on the Site Characterization Plan prepared by the DOE and has provided DOE with a significant list of issues to be resolved. This list is in the form of the Site Characterization Analysis issued by the NRC. The Commission should initiate inquiry about the importance to the function of NRC of having all of the issues and questions raised in the SCA resolved to the satisfaction of the NRC staff on a time schedule commensurate with licensing needs. Similar questions should be answered regarding the importance of having all study plans which are based on the contents of the SCP completed and submitted to the NRC staff before work on the associated topics is initiated.

4. The post-emplacement process for a repository involves a period during which the repository is to be monitored and for which retrieval of the waste is to be planned.

   a. There are no criteria for the thermal and other measurements that are to be made during this period. The Commission may want to explore the need for such criteria and, if found necessary, request the NRC staff to develop and promulgate them in order to ensure that technologies for data acquisition and interpretation can be provided in a timely fashion for the design of the EBS and the repository.

   b. The need to retrieve the waste after emplacement and back-filling influences the design of the repository and the EBS. The staff has not defined what type of retrieval will be required, the extent to which retrieval is likely to be needed, under what conditions retrieval is to be practiced, or the standards and criteria that would govern the retrieval. Owing to the importance of these issues to the design of the repository, the Commission should encourage the NRC staff to define more closely, prior to licensing, criteria for the various parts of the emplacement and retrieval process, the monitoring protocols that are expected to be applied by DOE, and the regulations that are needed for this part of the HLW disposal system. . . .

Impact of Long-Range Climate Change in the Southern Great Basin

The ACNW held meetings on the impact of long-range climate change in the Southern Great Basin in November 1992. Their report reads, in part:[2]

> The objective . . . of the meeting was to explore the state of knowledge of the potential impact of long-range climate change on the anticipated performance of the proposed high-level radioactive waste repository at Yucca Mountain, Nevada. The principal questions of concern to the Committee at this meeting were:
>
> — What is the significance of potential climate change in the Southern Great Basin to the integrity of the proposed HLW repository at Yucca Mountain?
>
> — What are the nature and quality of models that will be used for predicting the climate for the next 10,000 years at Yucca Mountain?
>
> — Are data and methods available to test and qualify the models?. . . .
>
> Presentations were made to the ACNW on: (1) the impact of climate change on the repository; (2) paleoclimatological and paleohydrological methodologies;. . . (3) the role and status of paleoclimatic and paleohydrologic data; and (4) the basis, role, and status of global and regional (southwestern U.S.) climate models. . . .

Several specific items came to our attention... that we believe are of sufficient importance and interest that they should be communicated to you. These include:

1. The current paleohydrologic and paleoclimatic studies at Yucca Mountain serve as a baseline for forecasting climate and for testing climatic models by hindcasting. These investigations will not be completed until late in this decade, at the earliest, thereby impeding timely analysis of the potential impact of climate change on the integrity of the proposed HLW site.

2. A critical element in determining the effect of climate change is the rate of infiltration (fracture and matrix permeability) through the vadose zone at Yucca Mountain. The relationship between precipitation and infiltration flux is an essential parameter in relating predicted climatic conditions to the impact on the proposed repository. The definition of this parameter, its variability, and the related uncertainties should be given high priority.

3. Preliminary estimates of the impact of climate change over the next 10,000 years at Yucca Mountain indicate that the proposed repository will remain above the water table. However, these predictions are based on climatic and hydrologic models that are preliminary in nature and are supported by an inadequate data base. Additional data acquisition and analytical studies are warranted. Sensitivity studies should be conducted to determine the degree of uncertainty that can be accepted in these data and these models without invalidating conclusions regarding the likely impact of climate change on the repository.

4. The meeting revealed an apparent lack of intra- and intercommunication among the several disciplines involved in climate study (e.g., hydrology and climate modeling). While individual researchers displayed a high degree of understanding of their own science and mission, they also displayed a lack of awareness of important information that could have come from other investigators.

5. Climatology is a significant discipline that needs to be represented within the areas of staff expertise available to the Commission. There is a need to monitor the Yucca Mountain climate change program and especially the climate modeling efforts of the DOE contractors.

6. Not all current DOE programs aimed at investigating climate change at Yucca Mountain are being performed under the study plan submitted to the NRC. . . .

GAO REPORT RECOMMENDS IMPROVEMENTS FOR MONITORING HANFORD CONTAMINANTS

The General Accounting Office (GAO) released a detailed report following its examination of the DOE's programs to monitor contamination at the Hanford Site waste disposal facility in southeast Washington State.[3]

The study, conducted between June 1991 and March 1992, is entitled *Nuclear Waste—Improvements Needed in Monitoring Contaminants in Hanford Soils (GAO/RCED-92-149)*. The main focus of the report is the contamination level in the vadose zone—the unsaturated soil layer above the groundwater table.

Background

According to the report, the Hanford Site, managed by the DOE Richland Field Office, has been generating billions of gallons of liquid waste since 1943. Most of the waste was released into nearly 300 disposal sites, including trenches, ponds, and cribs—underground structures that allow liquid waste to percolate into the soil (now inactive and awaiting cleanup). According to DOE, approximately 440 billion gallons of liquid waste was disposed of this way and another 65 million gallons of high-level mixed (radioactive and hazardous) waste is stored underground in 28 double-shell storage tanks (clustered into 5 tank farms) and in 149 single-shell storage tanks (clustered into 12 tank farms). The DOE believes that possibly 66 single-shell tanks may have leaked and estimated that as much as one million gallons of high-level mixed waste has leaked into the soil. The DOE also believes that these radioactive materials have in some cases already reached the Columbia River.

The Westinghouse Hanford Company, the DOE Hanford Site operations contractor, is responsible for Hanford's vadose zone programs. Two DOE organizations, the Tank Farms Project Office and the Environmental Restoration Division, and two Westinghouse groups handle the two principal vadose zone programs (leak monitoring and inactive site characterization). The Westinghouse Tank Farms Surveillance and Data Acquisition Group performs routine vadose zone monitoring to detect leaks from the single-shell tank farms and the 12 active liquid-waste disposal cribs.

Questions Surrounding Vadose Zone Programs and Technology

Because there is no clear requirement to monitor the vadose zone, DOE has not developed a strategy for addressing its various vadose zone activities. However, in October 1989 Westinghouse issued a *Groundwater Protection Management Plan* that required the development of a site-wide vadose zone monitoring program. But, as of May 1992, no such program had been funded. Although Westinghouse had drafted a plan covering the two largest disposal areas at Hanford, it did not include an overall approach to managing vadose zone activities, cost data, or timetables for follow-on program activities.

Current plans for cleaning up Hanford's 1 500 liquid-waste disposal sites rely heavily on drilling new wells and analyzing soil samples to characterize the level of contamination. A 200-foot well costs over $150 000, and a full analysis of soil samples every 5 feet runs another $200 000. In addition, the cost of spectral gamma analysis is about $2 400 per well. According to the report, vadose zone technology in existing and new wells instead of physical sampling could prove to be a large savings. A March 1992 Westinghouse study concluded that as much as $130 million could be saved by reductions in the number of samples required. Also, the study identified 1 200 wells that would not have to be drilled for a savings of $180 million.

Conclusions

According to the GAO report, tracking the movement of these contaminants with the use of vadose technology will be crucial to the success of DOE's cleanup effort. But currently DOE is not using vadose technology at a level that effectively protects the public health and the environment. According to the GAO, relying on inadequate funding and out-of-date, uncalibrated equipment, DOE has been unable to identify leak plumes from tanks or inactive cribs.

Recommendations

To improve the vadose zone monitoring effort, the report recommends that the Secretary of Energy direct the Manager of the DOE Richland Field Office to do the following: (1) Review and update current monitoring procedures. This effort should require periodic calibration of the monitoring probes, use of appropriate logging speeds, and correction of radiation measurements. (2) Develop and implement the vadose zone monitoring plan called for in Hanford's *Groundwater Management Protection Plan.* This plan should include (a) an integrated management approach; (b) a strategy for modernizing existing vadose zone equipment; (c) a timetable, which should be tied to Hanford's cleanup schedule, for acquiring equipment and implementing program improvements, such as the installation of the calibration models; and (d) an approach for tracking the migration of contaminants from the active and inactive liquid-waste disposal sites.

YUCCA MOUNTAIN SITE NOT QUAKE PRONE, SCIENTIST SAYS

The site of the nation's planned nuclear waste dump has not been violently shaken by an earthquake in at least 10 000 years, according to a scientist who looks for toppled boulders to identify earthquake-prone areas.[4]

The *Associated Press* reported that James Brune, a seismological laboratory director at the University of Nevada-Reno, believes that the rocks and boulders balancing precariously on cliffs, hillsides, and ridges around Nevada's Yucca Mountain indicate that the proposed site has not been subjected to any major quakes in at least that period of time.

Brune's study does not rule out the possibility of major tremors at Yucca Mountain but indicates they rarely produce strong ground motion. Brune presented his findings at the 1992 Fall Meeting of the American Geophysical Union.

During the past year Brune drove 10 000 miles in California and Nevada looking for standing boulders. He found none within roughly 12 to 15 miles of the epicenters of earthquakes that measured between six and eight in magnitude since the mid-1980s. At greater distances he found boulders balanced precariously.

At Yucca Mountain, rocks appear to have been standing for 10 000 years, the minimum age to form the so-called black varnish on their surfaces, Brune said. The dark mineral layer forms when boulders are exposed to weather after adjacent rocks erode away. Some scientists believe the varnish is as much as 10 000 years old, Brune said. Nevada disputes the study.

"There are a lot of rocks out there, and there's evidence that other rocks have been toppled over [by earthquakes] sometime in the past, and it appears it may be recent past," asserted geologist C. Johnson, technical programs administrator at the Nevada Agency for Nuclear Projects. Johnson said other research indicates faults at Yucca Mountain have produced quakes of at least magnitude six within the last 10 000 years.

Brune countered that his findings are consistent with other research showing that strong quakes occur only every several thousand years on any particular major fault in Nevada.

Brune's study was financed by DOE, but he said the evidence that big earthquakes rarely cause violent shaking at Yucca Mountain, "is simple and straightforward. Anybody can go out and verify it."

DOE BRIEFS NRC ON RADIOACTIVE WASTE MANAGEMENT; GIVES YUCCA UPDATE

In early October 1992 DOE's Office of Civilian Radioactive Waste Management (OCRWM) reported to the NRC on plans for the storage of radioactive waste and on the status of the site characterization work at Yucca Mountain.[5]

The NRC has been deeply concerned about the ever-growing stockpile of radioactive waste in the United States. Commissioner J. Curtiss asked OCRWM whether they would be legally obligated to accept spent nuclear fuel in 1998 even if a Monitored Retrievable Storage (MRS) Facility were not ready to receive it. At the October meeting, OCRWM Director J. W. Bartlett responded, stating that "the Department's obligation to begin accepting spent nuclear fuel in 1998 arises following commencement of facility operations." He added that "neither the statute as a whole nor the Standard Contract purports to obligate the Department to begin accepting spent nuclear fuel in the absence of an operating facility at which the spent fuel can be either stored or disposed of in the fashion contemplated by the Act."

Bartlett claimed that DOE had "made a great deal of progress" in all aspects of the program, most importantly at what they hope will be the future radioactive waste repository at Yucca Mountain in Nevada. One of the most recent significant accomplishments, Bartlett said, was the beginning of excavation for underground testing. In addition, drilling and coring of the first deep unsaturated zone borehole was begun. Because of concerns that Yucca would be a hazard during earthquakes, a Seismic Action Plan was developed following a June 29, 1992, earthquake that had its epicenter 12.5 miles from the site and measured 5.6 on the Richter scale.

Yucca Mountain Site Characterization Project Manager C. Gertz said that procurement for one large tunnel boring machine (TBM) was to begin in October 1992. If DOE received the TBM by November 1993, tunnel boring could begin as early as February 1994.

Aside from all the technical work revolving around drilling tunnels and gathering data, there was the basic question of safety. "It's important to drill holes and drill tunnels but we want to answer questions: 'Is it safe or is it not safe?' " Gertz questioned.

Others at DOE shared in the hesitancy. "Can we license such a site," asked DOE Under-secretary H. Pomrehn. "I think it's open. I can't quite answer yet."

Nuclear Regulatory Commission Commissioner I. Selin, however, was running out of patience. Selin termed the search for a depository for 40 tons of highly toxic and radioactive material one of the most important environmental issues of our time. "I can't think of anything more important than finding an answer to these issues," he concluded.

INTERNATIONAL PROGRESS IN WASTE DISPOSAL

Several nations are successfully developing disposal facilities for radioactive waste, according to information provided by the U.S. Council for Energy Awareness. For example, France's Centre de l'Aube facility received its first waste shipment in 1992. It is France's second facility for low- and intermediate-level waste and will gradually replace the first, the Centre de la Manche, which opened in 1969. L'Aube uses a fully automated, monitored disposal vault concept, the model for the proposed low-level waste facilities in Pennsylvania and North Carolina.[6]

Olkiluoto, Finland's first underground repository for low- and medium-level waste, is now operating. The facility consists of two large silos, one for each type of waste, excavated in granite bedrock several hundred feet below ground and connected by tunnel to the Olkiluoto nuclear power plant. A second facility is being excavated at the Loviisa nuclear power plant.

Japan opened its first repository for low-level radiation in December 1992 at Rokkasho in the Aomori district. The site's operator, Japan Nuclear Fuel Ltd., accepted its first delivery: 1 480 drums filled with toxic waste from a power station. The facility's storage capacity of 50 000 drums is to be quadrupled by 1998 and will eventually expand to as much as 3 million drums. Located in northern Japan, the complex also includes a uranium enrichment plant, and a reprocessing plant is scheduled to be built there.[7]

BAN ON DUMPING OF WASTES IN THE OCEANS DISCUSSED

Moves to ban dumping of all radioactive and industrial wastes at sea was at the top of the agenda when signatories of the London Dumping Convention met in London in November 1992. Delegates, meeting at the International Maritime Organization (IMO), also considered the long-term strategy of the convention that was signed 20 years ago and examined the alleged dumping of radioactive waste materials in the Arctic Ocean. (See the article "General Administrative Activities" in this issue of *Nuclear Safety* on ocean dumping by the former Soviet Union.) The proposal to outlaw the disposal of radioactive and industrial wastes is opposed, among others, by the United Kingdom, which earlier in 1992 won agreement for a 15-year moratorium on the dumping of nuclear waste in the northeast Atlantic rather than a permanent ban. Delegates at previous meetings of contracting parties to the London Dumping Convention have called for an end to such practices, and the recent meeting was to look at ways of making these bans formal and permanent, according to a statement issued by the IMO.[8]

NRC CHANGES REGULATIONS TO ALLOW COMPACTED WASTE TO RETURN TO REACTOR SITES

The NRC amended its regulations to allow nuclear reactor licensees to receive back at the reactor site low-level radioactive waste generated at the site but sent off site for compaction or incineration to reduce its volume.[9]

The NRC said the amendment was needed primarily because of changing circumstances surrounding the treatment, storage, and disposal of low-level radioactive waste at nuclear power plants. However, the amendment applies to all reactor (power and nonpower) licensees.

When the current operating licenses for nuclear power plants were issued, low-level radioactive waste was being sent directly off site for disposal in a low-level radioactive waste disposal facility. Therefore the operating license did not authorize nuclear power plant operators to receive nuclear material (including waste) except in the form of fuel for use in the reactor or in the form of sealed radioactive sources for analysis, calibration, or other special purposes or if the nuclear material was associated with radioactive apparatus or components, such as contaminated pumps or tools. Thus, under the current licenses, reactor licensees could send low-level radioactive waste off site to another licensee for treatment (such as compaction or incineration) but could not receive the treated waste back at the nuclear power plant site.

RESEARCHERS WORK TO PROVE LONG-LIVED WASTE CAN BE BURNED IN INTEGRAL FAST REACTOR

The Experimental Breeder Reactor II at Idaho National Engineering Laboratory in Idaho is currently running as the prototype of the Integral Fast Reactor (IFR). "The IFR is a next generation power system which burns metal fuel, uses liquid-metal coolant, has passive safety characteristics and burns its own nuclear waste as fuel," Argonne said.[10]

The IFR Process

When EBR-II fuel reaches its maximum burnup, the rods are pulled out of the reactor to the IFR's Fuel Cycle Facility (FCF) for reprocessing the spent fuel. The reactor and the FCF form a closed loop between fuel burning and reprocessing useful material from the HLW. (Fuel recycling was planned for 1993). Fuel recycling at the reactor site, a key IFR feature, greatly reduces the amount of long-lived radioactive wastes that must be buried in secure geologic repositories, Argonne claimed.

Argonne said EBR-II's original fuel recycling building was being renovated so as to achieve reduced air pressure inside so that no airborne contamination can leak out. In case of even a severe accident, such as a strong earthquake, areas that contain radioactive materials would suck air into a special filtration system to hold in any contamination.

The FCF is both a production and an experimental facility. Argonne said that the planned 4-year demonstration will, "provide operating and maintenance records and allow fine-tuning of the process through experimentation. It will also prove the economic potential and commercial feasibility of the IFR."

The IFR's fuel recycling recovers the 80 to 85% of the fuel that is not burned in its first pass in the reactor as well as burnable by-products, Argonne said. This process is repeated until essentially all the fuel is used to produce electricity and most of the long-lived fission products and transuranics are "burned" up by being fissioned.

Argonne said a new development will permit a component of the fuel, zirconium, to be used as the fuel-rod mold, which will reduce the amount of waste. Currently, fuel rods are cast in disposable molds, which have to be treated as radioactive waste.

Recycling Steps

As the IFR fuel is recycled, it goes through eight remotely controlled steps. First, bundles of spent-fuel rods are removed from the IFR, disassembled, and chopped into small pieces. The pieces go into an electrorefiner where most of the uranium, plutonium, and other long-lived transuranic radioactive materials are separated from the short-lived fission products, which cannot be reused as fuel.

Next, a cathode processor further separates the metal. A casting furnace then forms the recycled long-lived materials into new fuel rods.

After pyro-processing, fuel rods are cut and inspected, then loaded into new cladding. The welding and settling system makes the closure weld on the top of each new fuel rod to seal it into its stainless-steel cladding and makes another inspection. The rods are then rebundled for reuse in the reactor.

Inherent Safety Features

The IFR uses a combination of metal fuel and a liquid-sodium coolant rather than the ceramic oxide fuel and water coolant used in most existing power plants. "The IFR combination enables the laws of nature to provide passively safe characteristics," Argonne researchers said.

If the temperature rises excessively in the core, the IFR fuel rods naturally begin to move away from each other and thus slow the rate of fission reactions and reduce the activity of the reactor. Metal fuel also has a high thermal conductivity, which allows heat from the core to be transported very efficiently to the coolant, a large reservoir of liquid sodium, which can accommodate overheating conditions that might pose serious problems in other reactor types, Argonne researchers explained.

Unlike water, sodium does not need to be kept at a high pressure to remain liquid at temperatures suitable for operating a power reactor. It therefore continues to perform when pressure is reduced if a leak or some other accident occurs, Argonne said. Furthermore, if power to coolant pumps is lost, the pool of coolant continues to accept heat and circulate naturally by convection to transport heat from the core.

Burning Actinides

In addition to burning its own nuclear waste, the IFR may be able to burn used fuel from current commercial reactors. Researchers have demonstrated that actinides created by the current generation of power reactors can be fabricated into IFR fuel rods for "burning" in the reactor.

Researchers cast three full-length fuel pins containing the components of metal IFR fuel—uranium, plutonium, and zirconium—and added the actinides americium and neptunium. Two of these pins will be fissioning together with other fuel in the reactor, whereas one is being thoroughly analyzed, according to Argonne staff members.

"The first experimental hurdle is over—the fuel has been cast. Now researchers must wait while the in-reactor experiment is performed. Then the fuel will be examined to see how well the new mixture of fuel behaves," Argonne said.

NRC TO PERMIT ON-SITE INCINERATION OF CONTAMINATED WASTE OILS AT NUCLEAR POWER PLANT SITES

The NRC has amended its regulations to permit the on-site incineration of waste oils used in nuclear plants and contaminated with very small amounts of radioactive materials.[11]

Previously, utility operators of nuclear power plants had to dispose of contaminated waste oils at low-level radioactive waste disposal facilities. In a few cases, licenses were specifically amended by the NRC staff to permit on-site incineration.

The amended regulations will permit the on-site incineration of waste oils (petroleum-derived or synthetic oils used principally as lubricants, coolants, hydraulic or insulating fluids, or metalworking oils) that have been contaminated with small amounts of radioactive materials in the course of the operation or maintenance of a nuclear power plant.

Releases of radioactive effluents, including those from waste-oil incineration, are limited to "as-low-as-reasonably achievable" levels already specified in Appendix I to Part 50 of the Commissions regulations. In addition, a generic assessment prepared for the staff showed that the environmental impacts of waste oil incineration will be minimal, NRC said.

The amendment to Part 20 of the NRC's regulations became effective on Jan. 7, 1993.

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Operating Experiences

Edited by G. A. Murphy

Summary of Fuel Performance Annual Report for 1990

By C. L. Painter, J. M. Alvis, and C. E. Beyer[a]

Abstract: *This article summarizes NUREG/CR-3950 (PNL-5210, Vol. 8),* Fuel Performance Annual Report for 1990, published in November 1993. *This thirteenth annual report provides a brief description of fuel performance during 1990 in commercial nuclear power plants and an indication of trends. Brief summaries of fuel design changes, fuel surveillance programs, fuel operating experience, fuel failure trends and problems, and high-burnup fuel experience are provided in this article.*

This review is the eighth in a series of *Nuclear Safety* articles[1–7] that summarize the *Fuel Performance Annual Report* (NUREG/CR-3950) published by the U.S. Nuclear Regulatory Commission (NRC). The NRC prepares the report to provide an annual review regarding nuclear fuel design changes, fuel surveillance programs, operating experience, fuel-related problems, and high-burnup experience. The performance of nuclear fuel in U.S. commercial reactors during calendar year 1990 is summarized in this article. A more detailed account of the information presented in this article is contained in NUREG/CR-3950 (PNL-5210, Vol. 8).

FUEL DESIGN CHANGES AND SUMMARY OF FUEL SURVEILLANCE PROGRAMS

During 1990 fuel vendors continued to develop new designs to improve fuel performance and reliability for fuel used in both pressurized-water reactors (PWRs) and boiling-water reactors (BWRs). The primary causes of fuel failure were debris, corrosion, and fretting wear. Various surveillance programs implemented by the fuel vendors to monitor the performance of new fuels and their results are discussed. Information was not provided by Babcock & Wilcox Fuel Company (BWFC) for the 1990 report.

ABB Combustion Engineering Nuclear Fuel (ABB CENF) (PWRs)

No specific new design changes were noted by ABB CENF in their 1990 letter report.[8] However, ongoing research and development programs are discussed in Ref. 9, from which the following information was taken.

ABB CENF has developed experience with mixed erbia (Er_2O_3)–UO_2 fuel used to supplement reactivity control. ABB CENF has found that, by using erbia in a significant fraction of fuel rods, it is possible to keep the erbia concentrations low. ABB CENF believes that erbia has some advantages over gadolinia and boron as a burnable poison. Erbia has a smaller cross section than gadolinium, which results in a smaller effect on the energy

[a]Pacific Northwest Laboratory. This work was supported by the U.S. Nuclear Regulatory Commission under Contract DE-AC06-76RLO 1830, NRC FIN L-1864. Pacific Northwest Laboratory is operated for the U.S. Department of Energy by Battelle Memorial Institute.

distribution of the neutron flux and thus minimizes power peaking. The cross section for erbium is similar to that for boron, but it depletes more slowly and thus prevents the larger power changes that would occur if boron were used to provide the same moderator temperature coefficient necessary for longer fuel cycles. Two experimental programs have been developed for erbia testing by ABB CENF. One involves four lead fuel assemblies (LFAs)[a] with 0.9 wt % Er_2O_3 and 3.4 wt % enriched UO_2 fuel, which were fabricated in 1989 and loaded into Calvert Cliffs Unit 2. The other program involves another four LFAs that contain fuel pins comprised of 1.5 wt % Er_2O_3 and 3.65 wt % enriched UO_2, which were inserted in San Onofre Unit 2 in 1991.[9]

ABB CENF has also developed three different fuel modifications to prevent fuel damage as a result of debris fretting. One design uses smaller flow holes in the bottom nozzle to block more debris. Another design uses long, solid end caps extended between the bottom nozzle and the bottom spacer grid. Any debris that is caught in this portion of the core will more likely damage the end cap rather than the hollow, fuel-containing portions located above. The third design, referred to as GUARDIAN,™ incorporates a special bottom grid to trap and retain debris during no-flow conditions. According to ABB CENF, the GUARDIAN™ design blocks 93% of debris and retains 76% of the trapped debris during no-flow conditions.

Fuel surveillance programs developed by ABB CENF, including the two discussed previously, are summarized in Table 1. The performance programs currently in progress will provide hot cell evaluation of fuel and cladding with peak local burnup approaching 70 GWd/MTU.

General Electric Company (GE) (BWRs)

No new design changes were reported by GE in 1990. It has made an effort to improve its fuel performance by reducing fuel failures caused by pellet–cladding interaction (PCI) and crud-induced localized corrosion (CILC). These two failure mechanisms have been identified as the most significant in GE fuel. GE estimates that 94% of all GE 8 × 8 fuel failures are a result of PCI (~14%) and CILC (~80%).[12]

The PCI fuel-failure mechanism was first addressed by GE in 1979 with the introduction of a barrier cladding fuel design. This fuel design incorporates a zirconium lining on the inside of the Zircaloy-2 cladding. The cladding design alleviates stresses caused by the fuel expanding faster than the cladding during power ramping. As of December 1990, over 920 000 GE barrier fuel rods have operated for at least one cycle with no observed failures caused by PCI.[13]

Crud-induced localized corrosion, identified in 1979 as a failure mechanism, occurs under very specific conditions in plants equipped with copper alloy condenser tubes and filter demineralizer condensate cleanup systems. General Electric has developed out-of-reactor tests to determine the susceptibility of Zircaloy to in-reactor corrosion. Additionally, manufacturing processes have been developed to improve the corrosion resistance of Zircaloy at the start of the manufacturing process and to maintain this resistance throughout the manufacturing process. Hatch Unit 1 and Unit 2 each had six lead use assemblies (LUAs) inserted for operation in 1988 to test manufacturing variables, such as heat treatment, surface conditioning, and cladding material. Three assemblies were removed from each core in 1990 after one cycle of operation with exposures up to 13 GWd/MTU. Examinations revealed little or no nodular corrosion on the rods. The next examinations were due in 1991. These and other GE surveillance programs are summarized in Table 1.

Siemens Nuclear Power Corporation (SNP) (PWRs and BWRs)

The SNP manufactures fuel for both PWR- and BWR-type reactors. Siemens 9 × 9 fuel type for BWRs has been developed with several different configurations of water rods available. Current types have one, two, or five water rods, and a nine water-rod type is undergoing testing. The various configurations produce lower linear heat generation rates to reduce fission-gas release and the likelihood of damage caused by PCI. The SNP has used axial zoning of gadolinia to improve uranium use and cold shutdown margins.[14]

There have been several developments in SNP's 17 × 17 fuel type for PWRs to improve fuel performance and prevent some types of fuel failures. Baffle jetting is a problem that occurs in PWRs when water flow causes vibration of fuel rods and subsequent cladding failure. Low-cost fuel-rod clips have been developed by SNP to prevent baffle jetting and, according to SNP, have essentially eliminated the problem. Another improvement has

[a]Lead fuel assembly (LFA) [ABB CENF], lead use assembly (LUA) [GE], and lead test assembly (LTA) [BWFC] denote the same concept.

been the development of high thermal performance spacers to improve heat transfer, lower cladding temperatures, and improve the fuel's departure from nucleate boiling margin.[14]

A major area of development at SNP has centered around reducing damage to fuel from debris.[14] The SNP has tested two design changes for lower grid plates that will reduce fuel susceptibility to damage from debris. The first design incorporates small flow holes to trap particles. A large number of 6-mm-diameter holes (as opposed to the previous 11-mm-diameter hole size) were 15% more efficient at trapping debris (67% compared to 52%). A new lower tie plate developed by SNP has been determined to be 97% efficient at blocking all major types of debris. The tie plate uses a curved grid to eliminate straight flow paths, is only 2 mm wide at the curved portion of the grid, and does not increase hydraulic resistance.

Major SNP fuel surveillance programs are summarized in Table 1.

Westinghouse Electric Corporation (W) (PWRs)

Westinghouse Electric Corporation has implemented several design improvements over the past years. The problem of fuel damage caused by debris was addressed with the advent of the debris filter bottom nozzle (DFBN). The DFBN has smaller flow holes in the bottom nozzle to block debris more efficiently; however, it maintains the same pressure drop as earlier fuel designs. In 1990 the DFBN was used in at least one region of fuel in 34 of the 59 W fueled reactors.[10]

Westinghouse has also developed the VANTAGE 5 and 5H fuel designs. The VANTAGE fuel designs have several performance-enhancing features. The VANTAGE 5 designs have been used in 38 W fueled commercial reactors.[10] Burnable absorbers, a reconstitutable top nozzle, intermediate flow mixer grids, and improved fuel use are incorporated into the VANTAGE 5 fuel designs. Intermediate flow mixer grids enhance flow turbulence, which results in an increase in the departure from nucleate boiling margin.

Westinghouse is currently using ZIRLO™ cladding containing niobium in the VANTAGE design to provide better corrosion resistance. The first two assemblies with ZIRLO™ cladding were irradiated up to 21 GWd/MTU in their first cycle, and one was reinserted. The second cycle is expected to achieve 37 GWd/MTU in early 1991.[10] This and other W fuel surveillance programs are summarized in Table 1.

FUEL OPERATING EXPERIENCE

The total number of fuel assemblies that were in or had completed operation in the United States increased from about 110 500 in 1989 to about 114 450 at the end of 1990. Of these assemblies, 69 350 were used in BWRs and 45 100 were used in PWRs.[15] The total number of fuel rods supplied to the world by the five U.S. nuclear fuel vendors through 1990 was over 16.3 million (11.5 million for PWRs and 4.8 million for BWRs).[16]

The Institute of Nuclear Power Operations (INPO) Fuel Reliability Indicator (FRI) has been adopted by several fuel vendors to assess the overall performance of fuel rods. The FRI for PWRs is determined by normalizing the ^{131}I coolant activity level to a standard cleanup system flow rate (referred to as "uncorrected activity") and correcting for tramp uranium[a] (referred to as the "corrected activity" or FRI value). The industry average coolant ^{131}I activity is typically 1.2×10^{-3} μCi/g of ^{131}I for PWRs. FRI values for BWR plants are determined from fission gas release measurements taken at the steam jet air ejector. The industry median FRI value for BWRs is 99 μCi/s.

ABB CENF

Calendar year 1990 batch-averaged burnup data for ABB CENF fuel are shown in Table 2. The highest batch-averaged burnup at discharge in 1990 was 44 GWd/MTU at St. Lucie Unit 2. The highest batch-averaged burnup in-reactor was 44.8 GWd/MTU at Arkansas Unit 2. However, a batch-averaged burnup at discharge of 56.8 GWd/MTU for four assemblies was attained in 1988 at Calvert Cliffs Unit 1.[17]

ABB CENF burnup experience with all Zircaloy fuel assemblies is shown in Table 3. The total number of active and discharged ABB CENF assemblies as of Dec. 31, 1990, was 8 400 (31% in core and 69% discharged). The total number of ABB CENF fuel rods was 1 615 797 (34% in core and 66% discharged).[8]

The average corrected coolant ^{131}I activity reported by domestic PWR plants using ABB CENF fuel for the period 1987 to 1990 is illustrated in Fig. 1. The average plant activity at the close of 1990 was 0.0055 μCi/g, and the median was 0.0027 μCi/g. These values compare well with industry standards as reported by INPO.[8]

[a]Tramp uranium is finely divided uranium oxide particles suspended in the coolant or deposited on core surfaces.

Table 1 Major Fuel Surveillance Programs: Status Through 1990

Vendor	Fuel type[a]	Power plant	Number of planned (completed) operating cycles	Scheduled completion of program	Inspections to date
ABB Combustion Engineering Nuclear Fuel	14 × 14[b]	Calvert Cliffs 1	5(5)	Complete	5
	14 × 14[b]	Fort Calhoun	6(6)	Complete	4
	14 × 14[c]	Calvert Cliffs 1	5(5), Pt. 1	Complete	5
	14 × 14[c]	Calvert Cliffs 1	5(5), Pt. 2	1993[d]	5
	14 × 14[e]	Calvert Cliffs 2	3(0)	1997	0
	16 × 16[e]	Arkansas 2[f]	3(3)	Complete	3
	16 × 16[g]	Arkansas 2	3(3)	Complete	3
	16 × 16[h]	Arkansas 2	5(5)	1992[d]	5
	16 × 16[e]	St. Lucie 2	3(2)	Complete	1
	16 × 16[g]	Palo Verde 1	3(2)	Complete	3
	16 × 16[i]	Palo Verde 1	3(1)	1994	2
	16 × 16[i]	Palo Verde 3	3(0)	1997	0
	16 × 6[e]	San Onofre 2	2(0)	1995	0
	14 × 14[j]	Maine Yankee	12(12)	1991	3
General Electric	Barrier LUAs[k]	Quad Cities 1	7(6)		6
	Barrier LUAs[k]	Quad Cities 2	5(6)		
	1981 LUAs[l]	Browns Ferry 3	(1)		
	1983 LUAs[m]	Peach Bottom 3	3(2)	1991	2
	1984 LUAs[n]	Duane Arnold	4(3)		3
	1987 LUAs[o]	Hatch 1	3(2)	1991	2
	Corrosion performance[p]	Hatch 1 and 2	1(2)	1991	1
	1988 LUAs[q]	Cooper	(2)		2
	1989 LUAs[r]	Peach Bottom 2	(1)	1991	0
	GE11 LUAs	3 reactors	(1)	1991	0
Siemens Nuclear Power Corporation	15 × 15	Robinson 2	5(5)	Complete	3
	14 × 14	Prairie Island 2	3(3)	Complete	1
	8 × 8	Oyster Creek	5(5)	Complete	5
	11 × 11	Big Rock Point	4(4)	Complete	3
	14 × 14	Ginna	5(5)	1990	3
	17 × 17	Blayais 3	4(4)	1990	3
	8 × 8	WNP 2	4(4)	1991	3
	14 × 14	Calvert Cliffs	3(0)	1993	0
	15 × 15	Palisades	3(1)	1993	1
	9 × 9	Hatch 2	3(1)	1994	1
	9 × 9	Hatch 1	3(1)	1995	1
Westinghouse	s	North Anna 1	4(4)	Complete	
	17 × 17 (OFA- Demo)[t]	Farley 1	4(4)[u]	Complete	4[t]
			5(5)[v]	Complete	5
	17 × 17 (OFA- Demo)[t]	Salem 1	4(4)[v]	Complete	3
	17 × 17 (OFA- Demo)[t]	Beaver Valley 1	3(3)[w]	Complete	3
	14 × 14 (OFA- Demo)[x]	Point Beach 2	4(4)[y]	Complete	4[y]
	17 × 17 (VANTAGE-5 Demo)	Summer 1	3(3)[z]	Complete	1
	IFBA Demo fuel rods[aa]	Turkey Point 3	(2)		
	IFBA Demo fuel rods[bb]	Turkey Point 4	(2)		
	IFM Demo assembly[cc]	McGuire 1	(2)		
	DFBN assembly[dd]	3 Plants			
	ZIRLO-clad fuel rod assembly[ff]	North Anna 1	3(1)[ee]		
	MO_2[gg]	R. E. Ginna	4(4)[hh]		

Table 1 (Continued)

[a]LTA, lead test assembly; LUA, lead use assembly; MO_2, mixed oxide (UO_2–PuO_2) fuel; R, retrofit fuel design, D, demonstration; OFA-Demo, Demonstration Optimized Fuel Assembly; IFBA, integral fuel burnable absorber; IFM, intermediate flow mixer; FPIP, Fuel Performance Improvement Program; DFBN, debris filter bottom nozzle; ZIRLO, an advanced zirconium alloy cladding that contains niobium.

[b]Standard-design, high-burnup program.

[c]Standard and advanced fuel design LTAs.

[d]Hot cell examination of high-burnup fuel yet to be performed.

[e]Burnable poison irradiation program.

[f]Arkansas Nuclear One-Unit 2 (also known as ANO-2).

[g]Standard surveillance program.

[h]Standard and advanced fuel design, high-burnup program.

[i]Advanced cladding designs.

[j]Hot cell examination of high-exposure control element assembly.

[k]Four bundles with barrier cladding at Quad Cities 1 were involved. Cycles 6 and 7 involved six rods removed from the initial bundles and placed in another assembly for further irradiation. At Quad Cities 2, 144 barrier bundles were used, 16 of which continue to be irradiated in their sixth cycle.

[l]Eight bundles with improved design features were involved.

[m]Four bundles with improved design features were involved.

[n]Five bundles with improved design features were involved.

[o]Four bundles. Program objective: lead use GE 8 × 8 NB.

[p]Six fuel bundles each at Hatch 1 and 2. Program objective: cladding material process.

[q]Four fuel bundles. Program objective: lead use GE 8 × 8 NB-1 features.

[r]Four GE 8 × 8 NB bundles.

[s]Eight fuel assemblies were irradiated as part of an EPRI program for their fourth consecutive 18-month operating cycle; four of the eight were in relatively high-power positions and attained an assembly average burnup of about 58.1 GWd/MTU at discharge (May 1989); the LFA average burnup was 58.4 GWd/MTU.[10]

[t]Two OFA-Demo assemblies.

[u]The two OFA-Demo assemblies in Farley 1 and the two assemblies in Salem 1 were discharged in 1984 after four cycles for examination. Burnup achieved: 39.1 GWd/MTU in Farley-1 and 34.4 GWd/MTU in Salem 1.[11]

[v]One of the two OFA-Demo assemblies was reinserted for irradiation (fifth cycle) and achieved a burnup of 52.8 GWd/MTU.[11] One standard fuel assembly (the symmetric partner to the OFA-Demo assembly in Cycle 7) was also irradiated for a fifth cycle and attained an average burnup of 52.1 GWd/MTU.[11]

[w]The two assemblies achieved a burnup of 35.5 GWd/MTU,[11] were discharged in 1984 after three cycles, and were examined.

[x]Two assemblies.

[y]The four assemblies completed their second cycle of irradiation in 1983. Subsequent examination showed one assembly had nine failed fuel rods (cause: fretting wear at bottom Inconel spacer grid). The other three assemblies were in good condition, returned to the core for a third and fourth cycle of irradiation, discharged in 1985, and examined.[11] Average burnup achieved was 40.3 GWd/MTU.[11]

[z]The four assemblies completed their third cycle of irradiation and were discharged in 1988 after attaining an accumulated average burnup of 46.0 GWd/MTU.

[aa]The four IFBA rods were monitored during irradiation by in-core instrumentation.

[bb]There were 28 IFBA rods in each of four demonstration assemblies; this allowed removal of some of the rods for postirradiation examination.

[cc]One characterized IFM spacer grid demonstration assembly.

[dd]Three fuel assemblies with DFBNs.

[ee]The fuel rods attained a burnup of over 21.0 GWd/MTU in their first cycle, which was completed during February 1989. The rods are expected to surpass a burnup of 57.0 GWd/MTU at the completion of a third irradiation cycle.

[ff]Two demonstration fuel assemblies with ZIRLO-clad fuel rods began irradiation in June 1987. ZIRLO is an advanced zirconium alloy that contains niobium. ZIRLO is a trademark of Westinghouse Electric Corporation, Pittsburgh, Pennsylvania.

[gg]Four assemblies with W mixed oxide fuel rods were involved. The mixed oxide ($U0_2$–PuO) fuel rods for Ginna were manufactured by W, but their irradiation was not part of a W development program.

[hh]The four assemblies were irradiated for the fourth cycle (i.e., they were in the Cycle 11–14 cores) and were discharged. Average burnup was 38.5 GWd/MTU.

Table 2 Summary of ABB CENF Fuel Irradiated and/or Discharged in 1990

		Number of assemblies		Number of fuel rods		Batch-averaged burnup, GWd/MTU	
Reactor (fuel cycle)	Fuel batch	In reactor at end of year	Discharged during year	In reactor at end of year	Discharged during year	On Dec. 31, 1990	At discharge
Arkansas 2	F	17	0	4 012	0	44.8	
(cycle 8)	H	28	0	6 352	0	41.9	
	J	68	0	15 312	0	34.4	
	K	64	0	14 416	0	15.8	
Calvert Cliffs 1	K	69	0	12 144	0	33.5	
(cycle 10)	L	52	0	9 152	0	21.3	
	M	92	0	15 280	0	10.6	
Calvert Cliffs 2	H	69	0	12 144	0	43.0	
(cycle 8)[a]	J	60	0	10 560	0	34.0	
	K	88	0	14 800	0	22.0	
Fort Calhoun	M	41	3	7 048	504	31.8	33.0
(cycles 12 and 13)	N	44	0	7 552	0	19.1	
	P	40	0	6 784	0	5.5	
Maine Yankee	N	0	64	0	10 880		40.5
(cycles 11 and 12)	P	72	0	12 400	0	33.4	
	Q	72	0	12 464	0	21.5	
	R	72	0	12 448	0	5.2	
Palo Verde 1	B	1	96	220	21 120	25.0	30.0
(cycles 2 and 3)	C	52	12	12 016	2 704	27.0	34.0
	D	80	0	18 528	0	19.0	
	E	108	0	24 240	0	7.0	
Palo Verde 2	B	1	68	220	14 960	24.0	30.2
(cycles 2 and 3)	C	36	28	8 496	6 224	26.0	33.5
	D	108	0	24 400	0	18.0	
	E	96	0	21 616	0	6.0	
Palo Verde 3	A	0	69	0	16 284		15.3
(cycles 1 and 2)	B	73	35	16 060	7 700	27.0	17.6
	C	64	0	14 720	0	25.0	
	D	104	0	23 584	0	15.0	
St. Lucie 2	D	0	4	0	944		44.0
(cycles 5 and 6)	E	12	45	2 800	10 412	36.0	42.0
	F	49	27	11 380	6 156	32.0	34.0
	G	80	0	18 448	0	18.0	
	H	76	0	17 456	0	1.0	
San Onofre 2	A	1	0	236	0	21.0	
(cycle 5)	F	108	0	24 112	0	33.0	
	G	108	0	24 112	0	12.5	
San Onofre 3	A	1	5	236	1 180	15.0	31.0
(cycles 4 and 5)	D	0	16	0	3 776		30.5
	E	0	88	0	20 320		35.0
	F	108	0	24 112	0	27.5	
	G	108	0	24 112	0	5.5	
Waterford 3	C	1	0	224	0	34.6	
(cycle 4)	D	48	0	11 232	0	39.0	
	E	84	0	18 896	0	27.6	
	F	84	0	18 896	0	8.7	
Yankee Rowe	B	0	36	0	8 222		32.0
(cycles 20 and 21)	C	36	4	8 222	868	17.0	20.0
	D	40	0	9 090	0	1.3	

[a]Calvert Cliffs-2 did not operate during 1990.

Table 3 ABB CENF Burnup Experience with All-Zircaloy Assemblies Status as of December 31, 1990[a]

Fuel assembly batch burnup, GWd/MTU	Number of assemblies				Number of fuel rods			
	14 × 14	16 × 16	Other[b]	Total	14 × 14		Other[b]	Total
				In-Core Fuel Assemblies with Pressurized Fuel Rods				
0 to 3.999	0	76	40	116	0	17 456	9 090	26 546
4.000 to 7.999	112	312	0	424	19 232	69 968	0	89 200
8.000 to 11.999	92	0	0	92	15 280	0	0	15 280
12.000 to 15.999	0	361	36	397	0	81 244	8 222	89 466
16.000 to 19.999	44	160	0	204	7 552	39 976	0	44 528
20.000 to 23.999	212	109	0	321	36 416	24 636	0	61 052
24.000 to 27.999	0	66	0	66	0	15 160	0	15 160
28.000 to 31.999	41	282	0	323	7 048	63 568	0	70 616
32.000 to 35.999	201	296	0	497	35 104	66 816	0	101 920
36.000 to 39.999	0	13	0	13	0	3 024	0	3 024
40.000 to 43.999	69	76	0	145	12 144	17 584	0	29 728
44.000 to 47.999	0	17	0	17	0	4 012	0	4 012
48.000 to 51.999	0	0	0	0	0	0	0	0
52.000 to 55.999	0	0	0	0	0	0	0	0
56.000 to 59.999	0	0	0	0	0	0	0	0
Total	771	1 768	76	2 615	132 776	400 444	17 312	550 532
				Discharged Fuel				
0 to 3.999	0	0	0	0	0	0	0	0
4.000 to 7.999	6	0	0	6	1 048	0	0	1 048
8.000 to 11.999	97	0	136	233	16 148	0	28 752	44 900
12.000 to 15.999	247	387	72	706	42 935	91 220	15 284	149 439
16.000 to 19.999	256	192	0	448	44 344	44 448	0	88 792
20.000 to 23.999	151	104	8	263	25 276	23 392	1 728	50 396
24.000 to 27.999	476	478	0	954	81 034	107 256	0	188 290
28.000 to 31.999	795	326	100	1 221	136 552	73 080	22 090	231 722
32.000 to 35.999	536	471	36	1 043	93 660	109 320	8 222	211 202
36.000 to 39.999	222	210	0	432	39 008	48 296	0	87 304
40.000 to 43.999	316	151	0	467	54 474	34 988	0	89 462
44.000 to 47.999	0	4	0	4	0	944	0	944
48.000 to 51.999	2	1	0	3	349	230	0	579
52.000 to 55.999	1	0	0	1	176	0	0	176
56.000 to 59.999	4	0	0	4	702	0	0	702
Total	3 109	2 324	352	5 785	535 706	533 174	76 076	1 065 265

[a]From Ref. 8.
[b]ABB CENF or W 15 x 15 lattice with cruciform control blades (Palisades and Yankee Rowe).

ABB CENF estimates that 75% of the leaking fuel fabricated after 1983 (current fabrication process) and used between 1987 and 1990 failed because of debris-induced fretting of the Zircaloy-4 fuel cladding. Many of these leaking fuel rods were removed and replaced with nonfueled rods during refueling outages with the use of ABB CENF fuel assembly reconstitution methods. Overall, the reliability of ABB CENF fuel from 1983 to 1990, excluding failures caused by fabrication processes, is estimated to exceed 99.998% (Ref. 8).

General Electric Company

As of Dec. 31, 1990, over 4.0 million GE 8 x 8 fuel-type production Zircaloy-clad UO_2 rods were in or had completed operation in commercial BWRs. At the same time, over 1.5 million GE fuel rods were in operation.

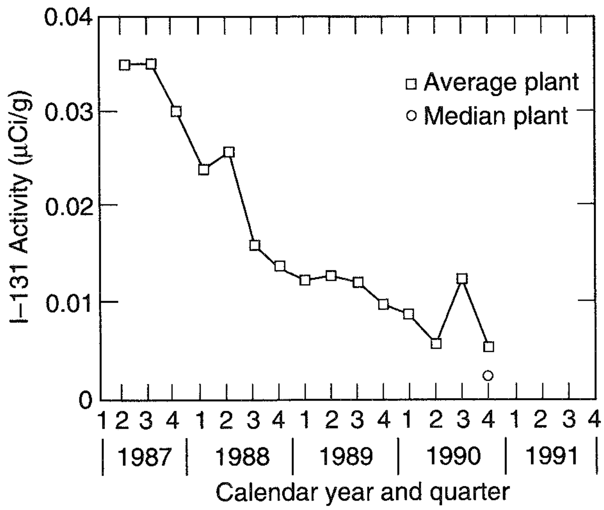

Fig. 1 Corrected coolant activity vs time for ABB CENF fuel.[8]

Table 4 General Electric 8 × 8 Fuel Performance (August 1990)[a]

	All 8 × 8	Zirconium liner 8 × 8
Date introduced in manufacturing	1973	1983
Cumulative fuel rods loaded	3 900 000	1 250 000
Fuel rod reliability,[b] %		
Including crud-induced localized corrosion failures	99.981	99.988
Excluding crud-induced localized corrosion failures	99.996	99.998

[a]From Ref. 12.

[b]Based on fuel rods completing at least one cycle of operation.

Approximately 1.37 million of these were PCI-resistant barrier fuel rods. All the fuel that GE produced in 1990 was barrier-type fuel.[13]

In 1990, 16 domestic and 6 overseas GE BWR plants containing GE fuel had refueling outages with over 3300 new GE 8 × 8 fuel bundles loaded. Nearly 80% of this new fuel loaded was GE's latest production design (GE 8 × 8EB and GE 8 × 8NB). GE has achieved more than 45 GWd/MTU bundle average burnup with its commercial BWR fuel. This equates to about 60 GWd/MTU peak pellet exposure.[13] The reliability of GE's 8 × 8 fuel, as of August 1990, is summarized in Table 4 (Ref. 12).

Siemens Nuclear Power Corporation

As of Dec. 31, 1990, fuel manufactured by SNP had been loaded into 50 commercial light-water reactors (LWRs) in the United States, Europe, and Asia, including 23 BWRs and 27 PWRs. SNP has also supplied fuel to the Loss of Fluid Test (LOFT) reactor. SNP fuel comprises a total of 18 412 fuel assemblies containing 2 199 446 fuel rods that have been irradiated. Of these, 64% of the assemblies were irradiated in BWRs and 36% in PWRs. SNP fuel experience is summarized in Table 5, and burnup distributions are shown in Fig. 2 (Ref. 18).

Siemens Nuclear Power Corporation BWR 9 × 9 and PWR 17 × 17 fuel assemblies reached new high burnups during 1990. The highest exposures reached by BWR 9 × 9 and PWR 17 × 17 fuel were 40.0 GWd/MTU at Gundremmingen-C in Germany and 46.4 GWd/MTU at Donald C. Cook Unit 2 in Michigan. The highest assembly-averaged burnups reached by SNP fuel to date are 52.1 GWd/MTU in the R. E. Ginna PWR in New York and 45.1 GWd/MTU in the Big Rock Point BWR in Michigan.[18]

Table 5 Summary of Siemens Nuclear Power Corporation Fuel Experience Through December 31, 1990

	Number of assemblies		Number of fuel rods		Maximum burnup, GWd/MTU	
Type	In core	Discharged	In core	Discharged	In core	Discharged
BWR	8 147	3 635	552 295	243 412	40.0	45.1[a]
PWR	2 172	4 458	488 226	915 514	52.1	52.1
Total	10 319	8 093	1 040 520	1 158 926		

[a]Average of extended burnup rods transferred to a new host fuel assembly.

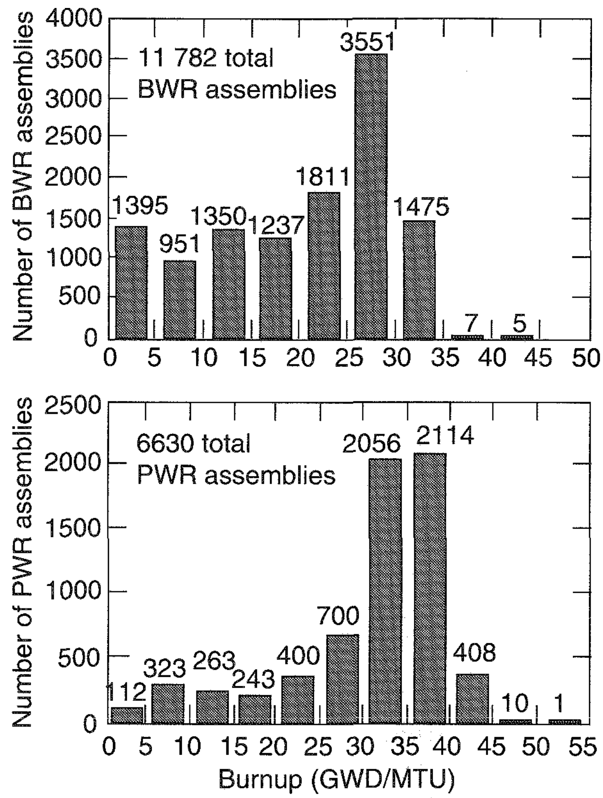

Fig. 2 Distribution of irradiated Siemens Nuclear Power Corporation fuel by assembly-averaged burnup through the end of 1990.[18]

SNP fuel reliability remained better than 99.997% in 1990; detailed failure statistics for SNP fuel rods are provided in Table 6. SNP has adopted the INPO FRI standard to assess fuel reliability. The FRI distribution for SNP PWR and BWR fuel is shown in Fig. 3. In 1990, SNP reported no fuel failures caused by design or manufacturing problems. Fuel failures caused by other than fuel design or manufacturing were determined to be the result of debris fretting. A 5-year trend in the FRI value for SNP fuel indicates continued improvement in fuel performance.[18]

Corrosion data were obtained by SNP at eight PWRs and four BWRs in 1990. Beta-quenched cladding reached exposures as high as 39.6 GWd/MTU and exhibited good resistance to corrosion in BWRs, particularly in those BWRs which are susceptible to CILC.

Westinghouse Electric Corporation

During 1990, 56 domestic commercial nuclear plants operated using W fuel. Approximately 2.65 million Zircaloy-clad fuel rods were in operation, which represented 10 760 fuel assemblies. Including discharged fuel, the number of irradiated W Zircaloy-clad fuel rods totals about 7.3 million, which represents 31 000 fuel assemblies.[10]

The average burnup of all W discharged fuel is about 29 GWd/MTU, and the average burnup of all W fuel (in-core plus discharged) is about 26 GWd/MTU. Burnup through the end of 1990 is summarized in Table 7. Assembly-averaged burnups in excess of 36 GWd/MTU have been achieved with 4 535 assemblies containing about 1.1 million rods. Of these, 1 264 assemblies with about 295 000 fuel rods reached burnups of over 40 GWd/MTU and 4 assemblies were irradiated to burnups of 55 GWd/MTU with a peak rod burnup of 60 GWd/MTU. Thirty-one W fueled plants have operated with fuel region average burnups in the range of 36 to 41 GWd/MTU, and coolant activities have remained low in these plants.

W reports its fuel reliability, accounting for all failure mechanisms, to be 99.998%. Uncorrected and corrected coolant activity level distributions for W fueled plants are shown in Table 8. As shown in the

Table 6 Siemens Nuclear Power Corporation Fuel-Rod Failure Statistics Through 1990

	Number of irradiated rods	Failed rods burnup less than warranted, fuel related		Failed rods burnup less than warranted, core related		All other SNP failures[a]		Total failures	
		Number	Rate, %	Number	Rate, %	Number	Rate, %	Number	Rate, %
BWR	795 706	49	0.006	103	0.013	14	0.002	166	0.021
PWR	1 403 740	9	0.001	130	0.009	70	0.005	209	0.015
Total	2 199 446	58	0.003	233	0.011	84	0.004	375	0.017

[a]Failures not examined and/or above warranted burnup.

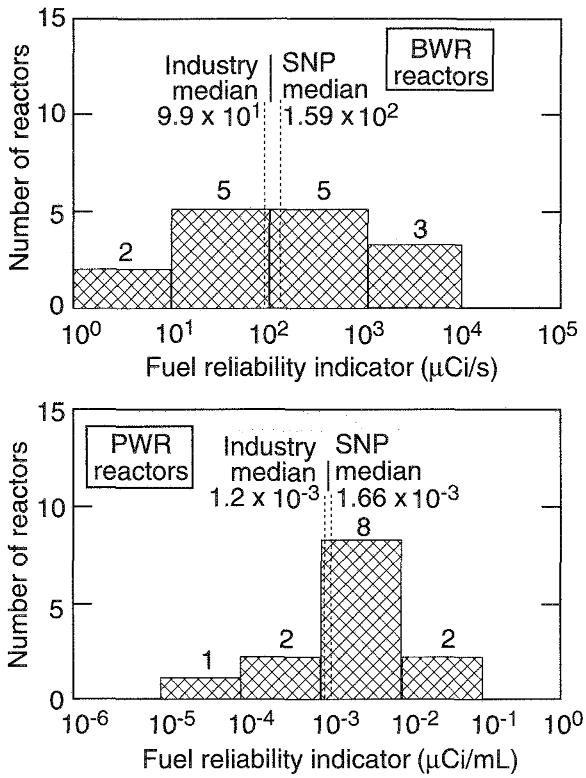

Fig. 3 Siemens Nuclear Power Corporation fuel reliability indicator (FRI) using the INPO standard method.[18]

table, 90% of W fueled plants have activity levels less than 0.01 μCi/g. The percentage of plants above the 0.03-μCi/g level has dropped from 38% in 1982 to 2% in 1990. The average uncorrected iodine activity has dropped from 0.041 μCi/g in 1982 to 0.0042 μCi/g in 1990 (one order of magnitude).

During 1990, ultrasonic testing examinations were performed at 21 reactor sites to identify leaking rods. In 65 assemblies at 19 sites, 82 leaking rods were identified. Fuel-assembly reconstitution was performed on 48 of the 65 assemblies. Of the 50 rods examined to date, 26 rod failures were caused by debris-induced fretting, 14 were caused by grid-rod fretting, 3 were caused by manufacturing-related causes, and 7 had no primary failure mechanism identified.

TRENDS REGARDING FUEL FAILURES AND FUEL-RELATED EVENTS

Over the past several years, considerable attention has been given to reducing the number of fuel failures in commercial reactors. Previously observed failure mechanisms are reasonably well understood today (e.g., PCI, CILC, debris fretting, and hydriding). As a result, in many instances utilities no longer consider common fuel failures to be "off-normal" events reportable to the NRC.

Table 7 Westinghouse Zircaloy-Clad Fuel Burnup and Total Rod Burnup Through 1990[a]

Assembly burnup, GWd/MTU	14 × 14 Rods	15 × 15 Rods	16 × 16 Rods	17 × 17 Rods	Total rods
0 to 4	10 740	41 132	0	175 560	227 432
4 to 8	19 257	8 976	3 760	189 816	221 809
8 to 12	44 824	12 440	12 455	261 360	331 079
12 to 16	39 678	74 356	40 420	489 456	643 910
16 to 20	77 435	139 271	22 090	693 528	932 324
20 to 24	73 965	122 323	7 990	381 480	585 758
24 to 28	142 955	146 570	23 500	591 383	904 408
28 to 32	167 024	311 271	32 195	677 424	1 187 914
32 to 36	244 460	303 620	28 905	668 644	1 245 629
36 to 40	120 789	162 915	16 920	473 215	773 839
40 to 44	30 967	56 854	2 115	149 424	239 360
44 to 48	5 728	16 853	0	24 552	47 133
48 to 52	0	200	0	6 072	6 272
52 to 56	0	816	0	528	1 344
56 to 60	0	0	0	1 056	1 056
Total	977 822	1 397 597	190 350	4 783 498	7 349 267

[a]From Ref. 10.

Table 8 Summary of Westinghouse Coolant Activity Through 1990

	Uncorrected[a] ^{131}I		Corrected[b] ^{131}I	
^{131}I activity range,[c,d] μCi/g	Number of plants in range	Percentage of plants in range	Number of plants in range	Percentage of plants in range
0.030 to 0.100	1	2	0	0
0.010 to 0.030	5	8	3	5
0.003 to 0.010	15	25	9	15
0.001 to 0.003	17	29	6	10
Below 0.001	21	36	41	70

[a]Uncorrected: normalized measured data.
[b]Corrected: normalized measured data corrected for tramp uranium.
[c]^{131}I values are given as of the end of 1990 (December basis).
[d]All data have been normalized to 100% power and the same cleanup rate.

Furthermore, some fuel vendors are not reporting the number or the cause of fuel failures in their fuel operating experience reports to NRC. Because neither the utilities nor the vendors are reporting the details regarding fuel failures, an accurate analysis of fuel-failure trends is difficult.

A recent Electric Power Research Institute paper[19] reported the failure percentages attributed to debris fretting, CILC, and PCI over the past 5 years. It shows an increase in the percentage of fabrication-oriented problems, grid fretting problems (PWRs), and unknown problems. These trends are illustrated in Table 9.

The largest percentage of fuel failures (48%) reported in 1990 were classified as unknown. This is of particular concern because new, previously unidentified, fuel-failure mechanisms may be grouped into this category. Problems labeled as "unknown cause" are significant and need to be addressed if fuel performance is to be improved further. Additional monitoring, inspection, data gathering, and studies are needed to correctly identify, model, and develop solutions for pin failure phenomena. Further studies may also improve the utilities' ability to better identify fuel-failure causes and then determine if modifications to plant operations and maintenance practices are needed to further reduce fuel failures.

Table 9 Failure Mechanisms Over the 1986–1990 Period[a]

Cause	1986–1987 percentage	1988–1989 percentage	1990 percentage
	PWRs		
Handling damage			2
Debris	9	54	14
Baffle-jetting	1	1	
Grid fretting	1	4	17
Primary hydriding	10	2	
Other fabrication	1	10	19
Other hydraulic	1	1	
Unknown	76	28	48
Total	100	100	100
	BWRs		
CILC	64	71	32
Fabrication	4	4	10
PCI	0	1	
Debris			10
Unknown	32	24	48
Total	100	100	100

[a]From Ref. 19.

Fuel-Related Events

In addition to fuel-failure mechanisms, other fuel-related problems continue to exist that directly affect the integrity of nuclear fuel. Major problem areas relate to fuel handling, control rod systems, and core–coolant problems (i.e., exceeding power levels and coolant flow limits and water chemistry specifications). Fuel-related events are reported to the NRC in accordance with the licensee event report (LER) system. A brief summary of each specific event is provided in NUREG/CR-3950 (PNL-5210, Vol. 8), *Fuel Performance Annual Report for 1990.* Figure 4 shows the number of reported events during 1990 for BWRs and PWRs. (NOTE: Only one fuel-failure event was reported to the NRC as an LER during calendar year 1990, yet several hundred fuel pins were known to fail during this same time period.)

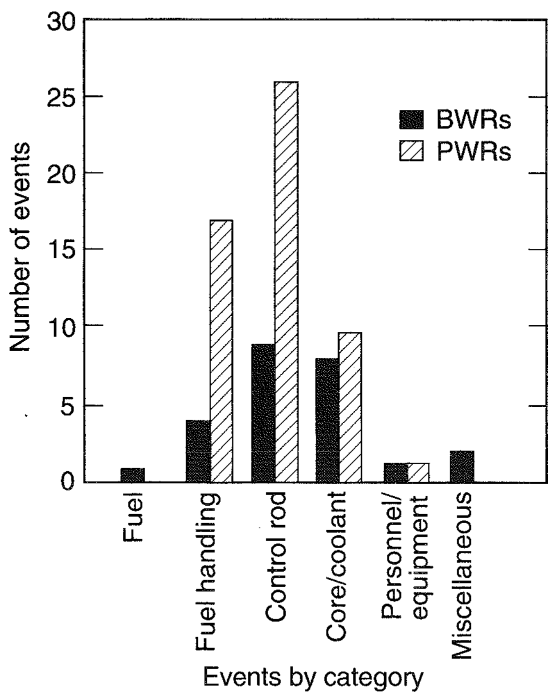

Fig. 4 BWR and PWR fuel-related events reported in 1990.

SUMMARY OF HIGH-BURNUP FUEL EXPERIENCE

In 1978, the Department of Energy (DOE) established an extended burnup program. The goal of the program was to demonstrate the technology necessary to extend discharge burnup levels to 45 GWd/MTU for BWRs and 50 GWd/MTU for PWRs.[20] Although reduction in fuel costs to utilities was the initial objective of the program, the industry has other reasons for extending fuel burnup, such as the reduction of generated spent fuel by LWRs (i.e., waste minimization) and improved capacity/availability factors.

The DOE estimated that, if target burnups of 50 GWd/MTU for PWRs and 45 GWd/MTU for BWRs were reached in all U.S. reactors, annual spent fuel generation would be reduced by 40% (Ref. 21). It is estimated that 26 commercial reactors will be required to expand their spent fuel storage capacity by the year 2000.[22] An industry-wide trend toward higher burnups may alleviate some spent fuel storage problems until a permanent and safe disposal facility can be constructed as required by the Nuclear Waste Policy Act of 1982.

Longer fuel cycles are beginning to be used in many plants today. The longer fuel cycles result in higher capacity/availability factors because the plant has less outage time. In most cases longer fuel cycles result in extended fuel burnups when compared to a standard 12-month cycle.

The 1978 extended burnup program resulted in newer fuel designs and improved manufacturing processes, which reduced many fuel-failure mechanisms that limited fuel performance. The program's burnup goals were partially achieved in 1982 by the discharge of two BWR assemblies at about 45 GWd/MTU and five PWR assemblies at about 50 GWd/MTU (Ref. 22). The extended burnup goals were further achieved in 1985 by the NRC's review and approval of vendor topical reports that addressed extended burnup experience, methodology, and tests. Although several utilities have achieved some extended burnup experience on whole fuel batches, the industry, on average, has yet to achieve high burnups. The highest annual average burnups for all BWR and PWR discharged assemblies were reached in 1990 (25.0 and 33.8 GWd/MTU, respectively). However, current burnup values for discharged fuel do not necessarily reflect the use of extended-burnup fuel currently in cores. Over the next several years this value will continue to rise as in-core extended-burnup fuel reaches its end of useful life.

In 1987, EPRI set a goal of 60 GWd/MTU assembly-averaged burnups to be reached by 1997.[23] These goals are being spurred by the industry's desire to achieve longer fuel cycles (18 to 24 months), which will reduce operating and fuel costs. Extending fuel burnup further will also continue to provide the following benefits: reduced costs associated with expanding spent fuel storage capacities and reduced uranium resource requirements. However, the lack of high-burnup data regarding cladding corrosion, ductility, fuel thermal conductivity, and the behavior of fuel and cladding during power transients and accidents will be a significant safety concern to the NRC as the industry seeks higher burnups.

The highest burnups achieved through 1990 by four of the nuclear fuel vendors are summarized in Table 10.

CONCLUSIONS

The average reliability of commercial nuclear fuel in the United States in 1990 was 99.998%. However, 48% of the reported fuel failures are classified as unknown.

Table 10 Highest Burnup Fuel Experience by Vendor

Vendor	Plant or test	Type	Burnup, GWd/MTU	Comment
ABB	ANO-2	PWR	44.8	Batch average[a]
CENF (Ref. 8)	St. Lucie 2	PWR	44.0	Batch average[b]
	702 rods discharged		56.0 to 59.9	Batch average[c]
GE (Ref. 11)		BWR	>45.0	Bundle average[e]
		BWR	60.0	Peak pellet exposure
SNP (Ref. 18)	R. E. Ginna	PWR	52.1	Assembly average[c]
	D. C. Cook 2	PWR	46.4	Assembly average[d]
	Big Rock Point	BWR	45.1	Assembly average[c,e]
	Gundremmingen-C	BWR	40.0	Assembly average[f]
W (Ref. 13)	Zion 1 and 2	PWR	55.0	Four assemblies average
			60.0	Peak rod burnup
	North Anna 1	PWR	58.4	Lead assembly average
			>60.0	Lead fuel rod average
BWFC		Information was not provided by BWFC		

[a]In core.
[b]Discharged in 1990.
[c]Highest to date.
[d]17×17 highest to date.
[e]Average of extended burnup rods transferred to a new host fuel assembly.
[f]9×9 highest to date.

The percentage of unidentified fuel failures are significant and may need to be addressed if fuel performance is to be improved further. Therefore a more comprehensive reporting system is needed to document fuel failures in the industry before an accurate analysis of fuel-failure trends may be made. Of the identified fuel failures, fuel vendors reported (on average) that debris-induced failures represent about 75%. Thus the driving force for the continued development of designs is intended to eliminate debris-induced fuel failures. Utilities will continue to seek higher fuel burnup to reduce fuel and operating costs and improve capacity/availability factors. In the long run, extended fuel burnup may reduce the costs associated with constructing additional spent fuel storage facilities.

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Reactor Shutdown Experience

Compiled by J. W. Cletcher[a]

This section presents a regular report of summary statistics relating to recent reactor shutdown experience. The information includes both numbers of events and rates of occurrence. It was compiled from data about operating events entered into the SCSS data system by the Nuclear Operations Analysis Center at the Oak Ridge National Laboratory and covers the three-month period of October, November, and December 1992. Cumulative information, starting from May 1, 1984, is also shown. Updates on shutdown events included in earlier reports are excluded.

Table 1 lists information on shutdowns as a function of reactor power at the time of the shutdown for both boiling-water reactors (BWRs) and pressurized-water reactors (PWRs). Only reactors in commercial operation at the start of the reporting period (October 1, 1992) are included. The second column for each reactor type shows the annualized shutdown rate for the reporting period. The third and fourth columns list cumulative data (numbers and rates) starting as of May 1, 1984.

Table 2 shows data on shutdowns by shutdown type: *Shutdowns required by Technical Specifications* are automatic scrams under circumstances where such a shutdown was required; *Intentional or required manual reactor protection system actuations* are manual shutdowns in which the operators, for reasons that appeared valid to them, took manual actions to actuate features of the reactor protection system; *Required automatic reactor protection system actuations* are actuations that the human operators did not initiate but that were needed; *Unintentional or*

Table 1 Reactor Shutdowns by Reactor Type and Percent Power at Shutdown[a] (Period Covered is the Fourth Quarter of 1992)

	BWRs (37)				PWRs (75)			
Reactor power (P), %	Number	Shutdown rate (annualized for period)	Cumulative number	Cumulative shutdown rate per reactor year[b]	Number	Shutdown rate (annualized for period)	Cumulative number	Cumulative shutdown rate per reactor year[c]
0	6	0.64	632	2.09	2	0.11	422	0.71
$0 < P \leq 10$	0	0.00	120	0.40	2	0.11	156	0.26
$10 < P \leq 40$	0	0.00	147	0.49	2	0.11	303	0.51
$40 < P \leq 70$	3	0.32	135	0.45	1	0.05	162	0.27
$70 < P \leq 99$	9	0.97	331	1.09	3	0.16	468	0.78
$99 < P \leq 100$	12	1.29	404	1.34	19	1.01	1009	1.69
Total	30	3.22	1769	5.85	29	1.54	2520	4.22

[a]Data include shutdowns for all reactors of the designated type while in commercial service during all or part of the period covered. The cumulative data are based on the experience while in commercial service since the starting date of Jan. 1, 1984, through the end of the reporting period; it includes the commercial service of reactors now permanently or indefinitely shut down.

[b]Based on cumulative BWR operating experience of 302.48 reactor years.

[c]Based on cumulative PWR operating experience of 597.51 reactor years.

[a]Oak Ridge National Laboratory.

Table 2 Reactor Shutdowns by Reactor Type and Shutdown Type[a] (Period Covered is the Fourth Quarter of 1992)

	BWRs (37)				PWRs (75)			
Shutdown (SD) type	Number	Shutdown rate (annualized for period)	Cumulative number	Cumulative shutdown rate per reactor year[b]	Number	Shutdown rate (annualized for period)	Cumulative number	Cumulative shutdown rate per reactor year[c]
SDs required by Technical Specifications	1	0.11	230	0.76	1	0.05	370	0.62
Intentional or required manual reactor protection system actuations	5	0.54	156	0.52	6	0.32	307	0.51
Required automatic reactor protection system actuations	17	1.82	824	2.72	20	1.06	1415	2.37
Unintentional or unrequired manual reactor protection system actuations	0	0.00	9	0.03	0	0.00	18	0.03
Unintentional or unrequired automatic reactor protection system actuations	7	0.75	550	1.82	2	0.11	410	0.69
Total	30	3.22	1769	5.85	29	1.54	2520	4.22

[a]Data include shutdowns for all reactors of the designated type while in commercial service during all or part of the period covered. The cumulative data are based on the experience while in commercial service since the starting date of Jan. 1, 1984, through the end of the reporting period; it includes the commercial service of reactors now permanently or indefinitely shut down.

[b]Based on cumulative BWR operating experience of 302.48 reactor years.

[c]Based on cumulative PWR operating experience of 597.51 reactor years.

unrequired manual reactor protection system actuations are essentially operator errors in which the human operators took action not really called for; and *Unintentional or unrequired automatic reactor protection system actuations* are instrumentation and control failures in which uncalled-for protective actuations occurred. Only reactors in commercial operation are included. The second column for each type of reactor shows the annualized rate of shutdowns for the reporting period. Cumulative information is shown in the third and fourth columns for each reactor type.

Table 3 lists information about shutdowns by reactor age category, both total numbers and rates in that category; it also shows cumulative results. Note that the age groups are not cohorts; rather reactors move into and out of the specified age groups as they age. The reactor age as used in this table is the number of full years between the start of commercial operation and the beginning of the reporting period (Oct. 1, 1992, for this issue). The first line of this table gives the information for reactors licensed for full power but not yet in commercial operation on that date.

Table 3 Reactor Shutdowns by Reactor Type and Reactor Age[a]
(Period Covered is the Fourth Quarter of 1992)

	BWRs (37)						PWRs (75)					
Years in commercial operation (C.O.)	Exposure during the period (in reactor years)	Number: Reactors	Number: Shutdowns	Shutdown rate (annualized for the period)	Cumulative number	Cumulative shutdown rate per reactor year	Exposure during the period (in reactor years)	Number: Reactors	Number: Shutdowns	Shutdown rate (annualized for the period)	Cumulative number	Cumulative shutdown rate per reactor year
Not in C.O.[b]	0.252	1	0	0.00	330	25.41	0.000	0	0	0.00	334	35.21
First year of C.O.	0.000	0	0	0.00	121	9.00	0.000	0	0	0.00	276	10.15
Second through fourth year of C.O.	0.252	1	1	3.97	260	6.34	1.051	5	6	5.71	501	5.55
Fifth through seventh year of C.O.	1.763	7	10	5.67	150	4.61	3.231	13	7	2.17	283	3.49
Eighth through tenth year of C.O.	1.511	6	4	2.65	170	6.19	2.771	11	1	0.36	340	4.30
Eleventh through thirteenth year of C.O.	0.000	0	0	0.00	269	5.89	1.763	7	4	2.27	472	4.50
Fourteenth through sixteenth year of C.O.	0.756	3	4	5.29	391	6.30	1.985	8	2	1.01	347	3.33
Seventeenth through nineteenth year of C.O.	2.267	9	4	1.76	250	5.18	4.517	19	6	1.33	202	2.79
Twentieth through twenty-second year of C.O.	1.763	7	6	3.40	112	5.19	2.535	12	3	1.18	57	2.51
Twenty-third through twenty-fifth year of C.O.	0.756	3	0	0.00	31	5.40	0.786	4	0	0.00	21	2.33
Twenty-sixth through twenty-eighth year of C.O.	0.000	0	0	0.00	8	2.67	0.000	0	0	0.00	12	4.00
Twenty-ninth through thirty-first year of C.O.	0.252	1	1	3.97	7	3.97	0.000	0	0	0.00	5	1.67
Thirty-second through ninety-ninth year of C.O.	0.000	0	0	0.00	0	0.00	0.252	1	0	0.00	0	0.00
Total	9.572		30	3.13	2099	6.65	18.891		29	1.54	2850	4.70

[a]Age is defined to be the time (in years) from the start of commercial operation to the time of the shutdown event, except for the first line, which lists reactors not yet in commercial service (see b below).

[b]This category includes reactors licensed for full-power operation but not yet commercial. During this reporting period reactors in this category included 1 BWR (Shoreham) and no PWRs.

Operating U.S. Power Reactors

Compiled by M. D. Muhlheim and E. G. Silver[a]

This update, which appears regularly in each issue of *Nuclear Safety*, surveys the operations of those power reactors in the United States which have been issued operating licenses. Table 1 shows the number of such reactors and their net capacities as of Dec. 31, 1992, the end of the three-month period covered in this report. Table 2 lists the unit capacity and forced outage rate for each licensed reactor for each of the three months covered in each report and the cumulative values of these parameters at the end of the covered quarter since the beginning of commercial operation. The information for this table was obtained from the Nuclear Regulatory Commission (NRC) Office of Information Resources Management. The Maximum Dependable Capacity (MDC) Unit Capacity (in percent) is defined as follows: (Net electrical energy generated during the reporting period × 100) divided by the product of the number of hours in the reporting period and the MDC of the reactor in question. The forced outage rate (in percent) is defined as (The total number of hours in the reporting period during which the unit was inoperable as the result of a forced outage × 100) divided by the sum (forced outage hours + operating hours).

Table 3 and Fig. 1 summarize the operating performance of the U.S. power reactors during the three months covered by this report (October, November, and December 1992) and for the years 1991 and 1992.

In addition to the tabular data, this article discusses other significant occurrences and developments that affected licensed U.S. power reactors during this reporting period. It includes, but is not limited to, changes in operating status, regulatory actions and decisions, and legal actions involving the status of power reactors. We do not have room here for routine problems of operation

Table 1 Licensed U.S. Power Reactors as of Dec. 31, 1992

Status	No.	Capacity,[a] MW(e) (net)
In commercial operation[b]	109	98 713
In power ascension phase[c]	0	0
Licensed to operate at full power	109	98 713
Licensed for fuel loading and low-power testing[d]	0	0

[a]Based on maximum dependable capacity (MDC) where available; design electrical rating (DER) is used when the MDC rating is not available.

[b]Excludes Dresden 1 (DER = 200), Fort St. Vrain (DER = 330), Humboldt Bay (DER = 65), LaCrosse (DER = 50), Rancho Seco (DER = 918), San Onofre 1 (DER = 436), Three Mile Island 2 (DER = 906), and Yankee Rowe (DER = 175), all of which have operating licenses but are shut down indefinitely or permanently.

[c]None at this time.

[d]None at this time.

[a]Oak Ridge National Laboratory.

Table 2 Summary of Operating U.S. Power Reactors as of Dec. 31, 1992[a]

Name and location (owner/operator)	Docket No.	Reactor type (reactor designer)	Design power		Commercial operation date	MDC unit capacity, %				Forced outage rate, %			
			MW(t)	MW(e)		Oct.	Nov.	Dec.	Cumulative (lifetime)	Oct.	Nov.	Dec.	Cumulative (lifetime)
ARKANSAS 1 and 2, Pope County, Ark.	50-313	PWR (B&W)	2568	850	12/74	99.9	101.0	101.1	60.7	0.0	0.0	0.0	11.8
(Arkansas Power & Light Co.)	50-368	PWR (CE)	2815	912	3/80	27.6	100.7	104.2	70.9	0.0	0.0	0.0	12.2
BEAVER VALLEY 1 and 2, Shippingport, Pa.	50-334	PWR (West)	2652	852	10/76	23.7	85.7	90.7	58.3	72.0	3.9	0.0	15.9
(Duquesne Light Co.)	50-412	PWR (West)	2660	836	11/87	102.1	93.7	88.1	76.6	0.0	0.0	0.0	3.2
BIG ROCK POINT, Charlevoix County, Mich. (Consumers Power Co.)	50-155	BWR (GE)	240	72	3/63	68.1	95.4	89.0	60.7	21.8	7.0	12.1	11.9
BRAIDWOOD 1 and 2, Braidwood, Ill.	50-456	PWR (West)	3425	1120	7/88	0.0	39.4	97.1	65.5	0.0	0.0	0.0	10.0
(Commonwealth Edison Co.)	50-457	PWR (West)	3425	1120	10/88	97.9	88.4	95.7	72.9	0.0	4.6	0.0	3.7
BROWNS FERRY 1, 2, and 3, Decatur, Ala.	50-259	BWR (GE)	3293	1065	8/74	0.0	0.0	0.0	31.1	100.0	100.0	100.0	59.2
(Tennessee Valley Authority)	50-260	BWR (GE)	3293	1065	3/75	92.6	84.1	74.8	36.7	0.0	0.0	0.0	50.2
	50-296	BWR (GE)	3293	1065	3/77	0.0	0.0	0.0	28.4	100.0	100.0	100.0	63.8
BRUNSWICK 1 and 2, Brunswick County, N. C.	50-325	BWR (GE)	2436	821	3/77	0.0	0.0	0.0	53.0	0.0	0.0	0.0	16.0
(Carolina Power & Light Co.)	50-324	BWR (GE)	2436	821	11/75	0.0	0.0	0.0	48.3	0.0	0.0	0.0	13.5
BYRON 1 and 2, Byron, Ill.	50-454	PWR (West)	3425	1120	9/85	96.7	96.5	91.5	72.5	0.0	0.0	0.0	2.6
(Commonwealth Edison Co.)	50-455	PWR (West)	3425	1120	8/87	98.5	98.7	92.4	69.6	0.0	0.0	0.0	3.0
CALLAWAY 1, Callaway County, Mo. (Union Electric Company)	50-483	PWR (West)	3411	1171	12/84	102.6	103.4	103.5	82.3	0.0	0.0	0.0	2.9
CALVERT CLIFFS 1 and 2, Lusby, Md.	50-317	PWR (CE)	2560	845	5/75	98.2	95.8	105.4	66.2	0.0	7.8	0.0	9.3
(Baltimore Gas & Electric Co.)	50-318	PWR (CE)	2560	845	4/77	90.9	102.8	103.4	69.6	9.4	0.0	22.9	5.9
CATAWBA 1 and 2, Lake Wylie, S. C.	50-413	PWR (West)	3411	1145	6/85	28.0	100.0	101.9	68.3	61.0	0.0	0.0	11.1
(Duke Power Co.)	50-414	PWR (West)	3411	1153	8/85	100.1	49.7	83.8	69.8	0.0	0.0	12.9	11.1
CLINTON 1, Clinton, Ill. (Illinois Power Co.)	50-461	BWR (GE)	2894	933	11/87	98.2	79.5	95.0	57.6	0.0	9.4	0.0	12.6
COMANCHE PEAK, Glen Rose, Tex. (Texas Utilities Electric Co.)	50-445	PWR (West)	3411	1150	8/90	58.5	0.0	2.0	61.7	6.2	0.0	9.1	7.9
COOK 1 and 2, Benton Harbor, Mich. (Indiana &	50-315	PWR (West)	3250	1030	8/75	2.1	86.7	99.6	66.0	22.4	0.0	0.0	6.4
Michigan Electric Co.)	50-316	PWR (West)	3391	1100	7/78	0.0	0.0	30.3	58.8	100.0	100.0	59.6	16.4

(Table continues on the next page.)

Table 2 (Continued)

Name and location (owner/operator)	Docket No.	Reactor type (reactor designer)	Design power MW(t)	Design power MW(e)	Commercial operation date	MDC unit capacity, % Oct.	Nov.	Dec.	Cumulative (lifetime)	Forced outage rate, % Oct.	Nov.	Dec.	Cumulative (lifetime)
COOPER, Nemaha County, Nebr. (Nebraska Public Power District)	50-298	BWR (GE)	2831	778	7/74	95.8	100.0	98.7	63.9	0.0	0.0	0.0	4.3
CRYSTAL RIVER 3, Crystal River, Fla. (Florida Power Corp.)	50-302	PWR (B&W)	2560	825	3/77	98.8	102.2	91.8	58.7	0.0	0.0	8.5	18.4
DAVIS-BESSE 1, Ottawa County, Ohio (Toledo Edison Co.)	50-346	PWR (B&W)	2772	906	7/78	100.3	100.3	100.8	50.9	0.0	0.0	0.0	22.9
DIABLO CANYON 1 and 2, Diablo Canyon, Calif. (Pacific Gas & Electric Co.)	50-275 50-323	PWR (West) PWR (West)	3338 3411	1086 1119	5/85 3/86	0.0 99.3	54.2 97.7	98.1 100.3	76.1 79.8	0.0 0.0	0.0 0.0	1.4 0.0	3.5 4.3
DRESDEN 2 and 3, Grundy County, Ill. (Commonwealth Edison Co.)	50-237 50-249	BWR (GE) BWR (GE)	2527 2527	794 794	6/70 11/71	86.3 67.4	84.2 71.7	44.3 75.3	58.2 56.1	0.0 26.3	0.0 10.8	32.4 11.9	12.1 11.3
DUANE ARNOLD, Cedar Rapids, Iowa (Iowa Electric Light & Power Co.)	50-331	BWR (GE)	1593	538	2/75	100.5	81.6	99.4	61.2	0.0	12.5	0.0	12.5
FARLEY 1 and 2, Dothan, Ala. (Alabama Power Co.)	50-348 50-364	PWR (West) PWR (West)	2652 2652	829 829	12/77 7/81	0.0 87.3	0.0 99.8	77.2 100.0	73.6 82.0	0.0 5.6	0.0 0.0	4.7 0.0	6.7 4.0
FERMI-2, Newport, Mich. (Detroit Edison Co.)	50-341	BWR (GE)	3292	1093	1/88	0.0	64.0	66.4	64.6	0.0	10.2	0.0	7.8
FITZPATRICK, Oswego, N. Y. (Power Authority of State of N. Y.)	50-333	BWR (GE)	2436	821	7/75	0.0	0.0	0.0	61.8	0.0	0.0	0.0	12.9
FORT CALHOUN, Washington County, Nebr. (Omaha Public Power District)	50-285	PWR (CE)	1420	478	6/74	101.0	101.5	101.7	68.5	0.0	0.0	0.0	4.3
GINNA, Ontario, N. Y. (Rochester Gas & Electric Corp.)	50-244	PWR (West)	1520	490	7/70	102.0	100.0	102.4	75.0	0.0	0.0	0.0	5.8
GRAND GULF 1, Port Gibson, Miss. (Mississippi Power & Light Co.)	50-416	BWR (GE)	3833	1250	7/85	102.6	104.1	105.4	75.9	0.0	0.0	0.0	6.2
HADDAM NECK, Haddam Neck, Conn. (Connecticut Yankee Atomic Power Co.)	50-213	PWR (West)	1825	582	8/67	102.7	103.7	103.9	75.0	0.0	0.0	0.0	5.7
HATCH 1 and 2, Baxley, Ga. (Georgia Power Co.)	50-321 50-366	BWR (GE) BWR (GE)	2436 2436	777 795	12/75 9/79	96.1 0.0	100.0 8.2	100.7 73.2	64.8 65.5	3.9 0.0	0.0 42.6	0.0 0.0	12.3 7.0
HOPE CREEK, Salem, N.J. (Public Service Electric & Gas Company)	50-354	BWR (GE)	3293	1067	12/86	0.0	63.6	93.6	80.1	0.0	0.0	6.9	5.0

Table 2 (Continued)

Name and location (owner/operator)	Docket No.	Reactor type (reactor designer)	Design power MW(t)	Design power MW(e)	Commercial operation date	MDC unit capacity, % Oct.	MDC unit capacity, % Nov.	MDC unit capacity, % Dec.	MDC unit capacity, % Cumulative (lifetime)	Forced outage rate, % Oct.	Forced outage rate, % Nov.	Forced outage rate, % Dec.	Forced outage rate, % Cumulative (lifetime)
INDIAN POINT 2 and 3, Buchanan, N.Y.													
(Unit 2, Consolidated Edison Co. of New York;	50-247	PWR (West)	2758	873	8/74	101.2	100.0	100.4	62.6	0.0	0.0	0.0	7.1
Unit 3, Power Authority of State of New York)	50-286	PWR (West)	2760	965	4/76	50.9	80.7	101.5	55.7	36.8	0.0	0.0	15.3
KEWAUNEE, Carlton, Wis. (Wisconsin Public Service Corporation)	50-305	PWR (West)	1650	535	6/74	102.3	96.1	102.7	82.4	0.0	5.0	0.0	2.3
LA SALLE 1 and 2, Seneca, Ill.	50-373	BWR (GE)	3323	1078	1/84	2.8	0.0	0.0	60.7	0.0	0.0	0.0	6.9
(Commonwealth Edison Company)	50-374	BWR (GE)	3323	1078	10/84	100.7	89.9	103.0	64.3	0.0	9.3	0.0	12.6
LIMERICK 1 and 2, Pottstown, Pa.	50-352	BWR (GE)	3293	1055	2/86	99.4	99.2	97.8	68.9	0.0	0.0	0.0	5.3
(Philadelphia Electric Company)	50-353	BWR (GE)	3293	1055	1/90	94.0	75.4	52.9	82.9	0.0	6.5	24.2	4.6
MAINE YANKEE, Lincoln County, Maine (Maine Yankee Atomic Power Company)	50-309	PWR (CE)	2560	790	12/72	96.9	101.3	86.1	71.9	0.0	0.0	13.5	7.5
McGUIRE 1 and 2, Cowans Ford Dam, N.C.	50-369	PWR (West)	3411	1180	12/81	98.6	100.0	100.5	61.6	0.0	0.0	0.0	13.5
(Duke Power Company)	50-370	PWR (West)	3411	1180	3/84	100.8	101.7	102.1	72.4	0.0	0.0	0.0	7.6
MILLSTONE POINT 1, 2, and 3, Waterford, Conn.	50-245	BWR (GE)	2011	660	3/71	98.3	97.5	92.5	69.6	0.0	0.0	2.6	12.5
(Northeast Nuclear Energy Company)	50-336	PWR (CE)	2560	870	12/75	0.0	0.0	0.0	64.3	0.0	0.0	0.0	15.5
	50-423	PWR (West)	3411	1150	4/86	0.0	61.5	75.2	67.4	100.0	28.1	0.0	18.8
MONTICELLO, Monticello, Minn. (Northern States Power Company)	50-263	BWR (GE)	1670	545	6/71	100.6	93.1	81.1	73.4	0.0	0.0	0.0	3.7
NINE MILE POINT 1 and 2, Oswego, N.Y.	50-220	BWR (GE)	1850	620	12/69	97.7	99.8	99.7	55.0	0.0	0.0	0.0	26.0
(Niagara Mohawk Power Corporation)	50-410	BWR (GE)	3323	1080	3/88	97.5	78.5	97.8	50.5	0.0	14.3	0.0	22.3
NORTH ANNA 1 and 2, Louisa County, Va.	50-338	PWR (West)	2775	907	6/78	77.7	58.8	46.4	65.7	0.0	0.0	0.0	11.4
(Virginia Electric & Power Company)	50-339	PWR (West)	2775	907	12/80	99.1	99.4	99.6	76.6	0.0	0.0	0.0	5.8
OCONEE 1, 2, and 3, Oconee County, S.C.	50-269	PWR (B&W)	2568	887	7/73	84.4	99.7	5.7	70.5	12.8	0.0	0.0	10.9
(Duke Power Company)	50-270	PWR (B&W)	2568	887	9/74	63.4	101.2	101.8	71.5	33.4	0.0	0.0	9.2
	50-287	PWR (B&W)	2568	887	12/74	53.3	100.3	102.0	71.3	42.8	0.0	0.0	11.0
OYSTER CREEK, Oyster Creek, N.J. (Central Power & Light Company)	50-219	BWR (GE)	1930	650	12/69	99.4	89.4	0.0	56.4	0.0	0.0	0.0	11.0
PALISADES, Covert Township, Mich. (Consumers Power Company)	50-255	PWR (CE)	2200	805	12/71	101.6	78.7	108.5	51.1	5.5	26.1	0.0	30.4

(Table continues on the next page.)

Table 2 (Continued)

Name and location (owner/operator)	Docket No.	Reactor type (reactor designer)	Design power		Commercial operation date	MDC unit capacity, %				Forced outage rate, %			
			MW(t)	MW(e)		Oct.	Nov.	Dec.	Cumulative (lifetime)	Oct.	Nov.	Dec.	Cumulative (lifetime)
PALO VERDE 1, 2, and 3, Wintersburg, Ariz.	50-528	PWR (CE)	3817	1270	2/86	93.8	102.5	92.5	55.2	4.8	0.0	6.6	18.1
(Arizona Public Service Company)	50-529	PWR (CE)	3817	1270	9/86	100.4	91.5	101.5	69.8	0.0	6.8	0.0	6.5
	50-530	PWR (CE)	3817	1270	1/88	0.0	8.2	103.8	69.2	0.0	0.0	0.0	8.0
PEACH BOTTOM 2 and 3, York County, Pa.	50-277	BWR (GE)	3293	1065	7/74	0.0	0.0	39.3	51.6	0.0	0.0	5.9	14.5
(Philadelphia Electric Company)	50-278	BWR (GE)	3293	1065	12/74	38.3	61.7	97.5	53.6	51.9	30.8	1.6	12.6
PERRY 1, Perry, Ohio (Cleveland Electric Illuminating Company)	50-440	BWR (GE)	3579	1205	11/87	74.1	97.4	100.2	69.9	24.7	0.0	0.0	8.0
PILGRIM 1, Plymouth, Mass. (Boston Edison Company)	50-293	BWR (GE)	1998	655	12/72	72.8	13.7	65.2	49.0	0.0	12.9	25.8	12.3
POINT BEACH 1 and 2, Manitowoc County, Wis.	50-266	PWR (West)	1518	497	12/70	97.1	101.0	101.7	75.4	3.4	0.0	0.0	1.7
(Wisconsin–Michigan Power Company; Wisconsin Electric Power Company)	50-301	PWR (West)	1518	497	12/72	0.0	35.6	100.5	81.7	0.0	0.0	0.0	1.1
PRAIRIE ISLAND 1 and 2, Red Wing, Minn.	50-282	PWR (West)	1650	530	12/73	47.2	0.0	0.0	81.4	29.8	0.0	0.0	5.4
(Northern States Power Company)	50-306	PWR (West)	1650	530	12/74	76.9	0.0	0.0	84.9	0.0	0.0	0.0	2.8
QUAD CITIES 1 and 2, Rock Island, Ill.	50-254	BWR (GE)	2511	789	2/73	0.0	0.0	36.8	65.1	0.0	0.0	0.0	5.8
(Commonwealth Edison Company)	50-265	BWR (GE)	2511	789	3/73	8.8	98.2	99.9	64.3	0.0	0.0	0.0	7.8
RIVER BEND 1, St. Francisville, La. (Gulf States Utilities Company)	50-458	BWR (GE)	2894	934	6/86	92.6	67.1	69.9	63.6	0.0	12.3	26.3	10.5
ROBINSON 2, Hartsville, S. C. (Carolina Power & Light Company)	50-261	PWR (West)	2200	700	3/71	103.3	101.3	107.3	62.5	0.0	0.0	0.0	14.8
SALEM 1 and 2, Salem, N.J.	50-272	PWR (West)	3423	1090	6/77	76.6	90.4	81.4	57.2	21.9	3.7	14.9	21.5
(Public Service Electric & Gas Company)	50-311	PWR (West)	3423	1115	10/81	100.1	87.8	100.1	56.6	0.0	8.5	0.0	22.9
SAN ONOFRE 1, 2, and 3, Camp Pendleton, Calif.	50-206	PWR (West)	1347	436	1/68	83.1	78.9	0.0	0.0	0.0	0.0	0.0	0.0
(Southern California Edison Company)	50-361	PWR (CE)	3410	1070	8/83	100.0	101.8	99.4	71.2	0.0	0.0	0.0	6.9
	50-362	PWR (CE)	3410	1080	1/84	101.5	99.6	102.4	71.4	0.0	0.0	0.0	6.9
SEABROOK 1, Seabrook, N.H. (Public Service Company of New Hampshire)	50-443	PWR (West)	3411	1150	8/90	0.0	35.3	87.9	74.3	0.0	7.5	8.3	5.5
SEQUOYAH 1 and 2, Daisy, Tenn.	50-327	PWR (West)	3423	1148	7/81	83.3	92.9	100.1	50.3	16.8	5.3	0.3	38.5
(Tennessee Valley Authority)	50-328	PWR (West)	3423	1148	6/82	88.2	97.5	97.9	54.0	0.0	0.0	0.3	33.2

Table 2 (Continued)

Name and location (owner/operator)	Docket No.	Reactor type (reactor designer)	Design power MW(t)	Design power MW(e)	Commercial operation date	MDC unit capacity, % Oct.	Nov.	Dec.	Cumulative (lifetime)	Forced outage rate, % Oct.	Nov.	Dec.	Cumulative (lifetime)
SHEARON HARRIS, Bonsal, N.C. (Carolina Power & Light Company)	50-400	PWR (West)	2775	900	1/87	0.0	0.0	89.9	74.9	0.0	0.0	0.0	4.1
SOUTH TEXAS 1 and 2, Bay City, Tex. (Houston Lighting & Power Company)	50-498 50-499	PWR (West) PWR (West)	3800 3800	1250 1250	8/88 6/89	0.0 99.2	0.0 95.7	0.1 90.7	62.0 69.7	0.0 0.0	100.0 0.0	97.7 6.6	14.4 12.0
ST. LUCIE 1 and 2, Hutchinson's Island, Fla. (Florida Power & Light Company)	50-335 50-389	PWR (CE) PWR (CE)	2560 2560	830 830	12/76 6/83	101.0 101.4	100.1 77.5	99.7 62.2	76.2 83.9	0.0 0.0	0.0 22.6	0.0 37.1	4.1 5.4
SUMMER 1, Broad River, S.C. (South Carolina Electric & Gas Company)	50-395	PWR (West)	2775	900	1/84	100.5	100.7	100.8	73.1	0.0	0.0	0.0	5.9
SURRY 1 and 2, Surry County, Va. (Virginia Electric & Power Company)	50-280 50-281	PWR (West) PWR (West)	2441 2441	788 788	12/72 5/73	99.1 98.5	100.1 99.0	99.9 98.6	60.0 60.1	0.0 0.0	0.0 0.0	0.0 0.0	18.4 14.4
SUSQUEHANNA 1 and 2, Berwick, Pa. (Pennsylvania Power & Light Company)	50-387 50-388	BWR (GE) BWR (GE)	3293 3293	1065 1065	6/83 2/85	99.8 0.0	90.7 41.8	98.1 101.9	72.6 77.8	0.0 0.0	7.9 5.5	0.0 0.0	7.6 5.4
THREE MILE ISLAND 1, Three Mile Island, Pa. (GPU Nuclear Corporation)	50-269	PWR (B&W)	2772	906	12/78	102.6	96.6	104.3	51.5	0.0	0.0	0.0	41.8
TROJAN, Columbia, Oreg. (Portland General Electric Company)	50-344	PWR (West)	3411	1130	5/76	95.8	26.8	0.0	54.2	0.0	72.3	100.0	13.7
TURKEY POINT 3 and 4, Dade County, Fla. (Florida Power & Light Company)	50-250 50-251	PWR (West) PWR (West)	2200 2200	693 693	12/72 9/73	0.0 17.3	0.0 93.5	73.7 103.2	60.4 60.7	0.0 2.0	0.0 0.0	0.0 0.0	12.5 12.0
VERMONT YANKEE, Vernon, Vt. (Vermont Yankee Nuclear Power Corporation)	50-271	BWR (GE)	1593	514	11/72	97.1	91.2	102.4	74.2	0.0	0.0	0.0	5.2
VOGTLE 1 and 2, Waynesboro, Ga. (Georgia Power Company)	50-424 50-425	PWR (West) PWR (West)	3411 3411	1157 1157	6/87 5/89	101.2 100.7	101.5 100.7	102.1 81.3	81.9 83.2	0.0 0.0	0.0 0.0	0.0 0.0	6.3 1.9
WASHINGTON NP 2, Richland, Wash. (Washington Public Power Supply System)	50-397	BWR (GE)	3323	1100	12/84	99.8	86.9	102.7	57.4	0.0	0.0	0.0	13.3
WATERFORD 3, Taft, La. (Louisiana Power & Light Company)	50-382	PWR (CE)	3410	1104	9/85	0.0	63.2	99.7	79.1	0.0	0.0	0.0	3.9
WOLF CREEK 1, Burlington, Kans. (Kansas City Power & Light Company)	50-482	PWR (West)	3411	1170	9/85	101.6	94.1	101.7	75.6	0.0	4.3	0.0	5.2
ZION 1 and 2, Zion, Ill. (Commonwealth Edison Company)	50-295 50-304	PWR (West) PWR (West)	3250 3250	1040 1040	12/73 9/74	85.5 71.3	95.6 19.7	98.8 0.0	56.3 60.7	7.1 0.0	0.0 0.0	0.0 0.0	16.7 15.4

[a]The information in this table is obtained from NRC Publication NUREG-0020, Vol. 16, Nos. 11 and 12, and Vol. 17, No. 1.

Table 3 Power Generation During the Fourth Quarter of 1992

Power generation	1990	1991	October	November	December	Year-to-date
Gross electrical, MW(e)h	605 169 082	643 414 027	51 437 865	53 232 735	61 004 990	650 121 847
Net electrical, MW(e)h	575 991 274	613 003 218	48 579 204	50 777 350	58 270 302	619 832 541
Average unit factors, %						
Service	71.1	73.6	69.8	75.3	81.9	74.6
Availability	71.1	73.6	70.5	75.3	81.9	74.8
Capacity						
MDC	67.0	70.2	66.9	71.6	79.2	71.3
DER	65.5	68.6	65.5	70.0	77.5	69.7
Forced outage rate	9.7	11.0	8.4	7.3	6.7	10.5

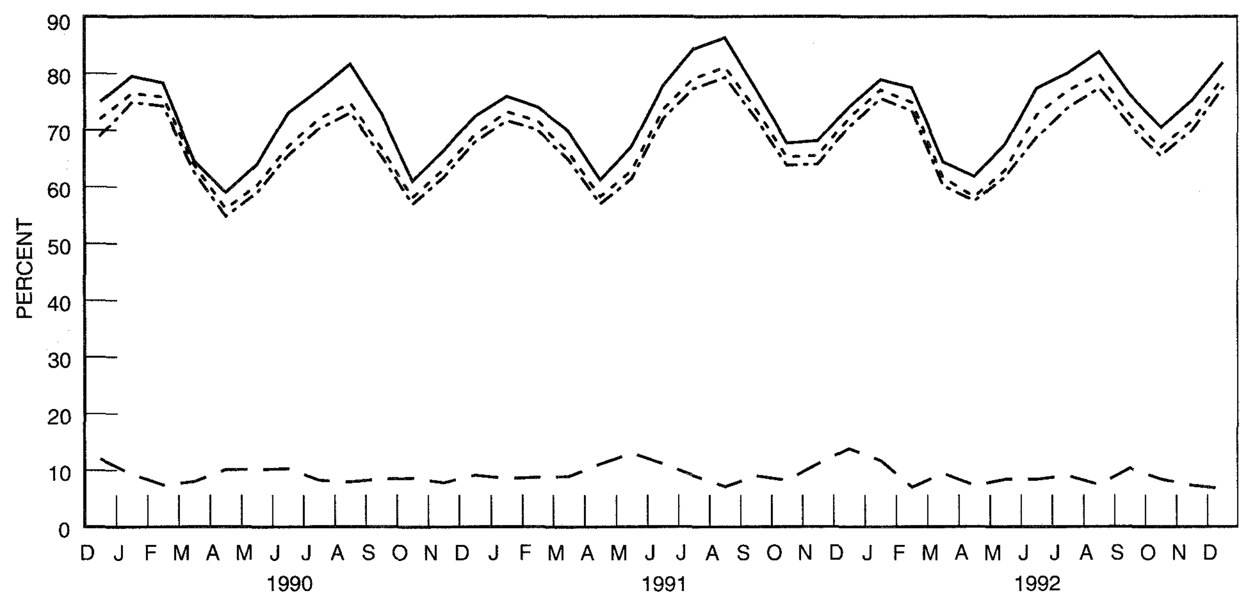

Fig. 1 Average unit availability, capacity factors, and forced outage rate. ——, availability factor. ...., MDC capacity factor. . — . —, DER capacity factor. — — —, forced outage rate. Data through February 1990 were obtained from the hard-copy version of NUREG-0200; data for the remainder of 1990 were obtained from the NRC Office of Information Resources Management. 1991 and 1992 data were obtained from the magnetic-media version of NUREG-0200.

and maintenance, but such information is available at the NRC Public Document Room, 2120 L Street, NW, Washington, DC 20555.

Some significant operating events are summarized elsewhere in this section, and, when appropriate, a report on activities relating to facilities still in the construction process is given in an article "Status of Power-Reactor Licensing Activities" in the last section of this journal. The reader's attention is also called to the regular features "General Administrative Activities," which deals with more general aspects of regulatory and legal matters, and "Waste and Spent Fuel Management," which covers legislative, administrative, and technical matters related to the back end of the fuel cycle and to management of radioactive wastes in general.

AEOD BRIEFS NRC ON 1991 REACTOR SAFETY ISSUES

Are the reactor safety standards that formed such an important part of the nuclear movement of the 1980s now taking a back seat to economic concerns? The NRC Office for Analysis and Evaluation of Operational Data addressed these questions, among other issues, in a briefing with NRC Commissioners in late September 1992.[1]

R. L. Spessard of the Division of Operational Assessment raised concerns over the fact that safety performance, which had steadily increased throughout the 1980s, had reached a plateau, perhaps associated with an increase of complacency in the nuclear industry. He noted that there are few industry safety initiatives but many more communications for information between the NRC and the industry instead of action. The AEOD staff questioned whether the complacency they believe exists is related to a shift from safety concerns to economic emphasis.

The NRC Commissioners seemed somewhat less certain that this plateau in safety was an altogether unfavorable happening. Commissioner J. R. Curtiss asserted that the steady improvements made during the past decade could quite possibly have brought the industry to an acceptable level of safety. Commissioner F. J. Remick concurred. "There will always be equipment failures. There will always be personnel failures," he said. "But have we reached the point where these are acceptable?"

The AEOD also listed the trends in several other safety categories from 1987 to 1991. In 1987 the average number of scrams per reactor was 4.25 per year. That figure had dropped to 2.25 by 1991, with projections for 1992 figured to be about the same. Equipment failures continued to be the principal cause of scrams.

Actual safety system failures dropped consistently from 1987 to 1991. In 1987 the average per reactor was 2.5, but it dropped to about 1.5 in 1990 and 1991. The radiation exposure rate for workers in the nuclear field remained relatively constant from 1986 to 1990. In 1986 the overexposure per 1000 workers had been 0.04%, but by 1990 it had dropped to 0.02%. The only jump came in 1990 when the number rose to 0.14%.

Other safety categories in which problems have diminished were safety-system actuations, automatic reactor trips, significant outages, and equipment-caused forced outages.

Spessard also identified various incident investigation team (IIT) and augmented investigation team (AIT) lessons learned at U.S. reactors. At Nine Mile Point in New York, the IIT noted potential generic design vulnerabilities, inadequacy of emergency procedure guidelines and training for the event of a loss of all annunciators, and concerns about the adequacy of control room staffing during emergencies. The AIT lessons at Vogtle (South Carolina), Oconee (South Carolina), and Oyster Creek (New Jersey) identified personnel errors and equipment problems. At Salem (New Jersey) the AIT discovered that multiple solenoid-operated valve failures were causing failure of the main turbine. At Diablo Canyon (California) and Palo Verde (Arizona), inadequate control over site vehicular traffic was discovered, and at FitzPatrick (New York) AIT concerns included an unmonitored release of radioactive steam from an auxiliary boiler vent. "The lessons [from these inspections] aren't being learned very well by the industry," Spessard said.

AVERAGE RADIATION DOSE TO WORKERS DIMINISHES

The average radiation dose per reactor to workers at U.S. nuclear power plants declined by 24% from 1991 to 1990, the NRC staff said. A report on "LWR Occupational Dose Data for 1991" showed that the average collective radiation dose per reactor for 1991 was 253 person-rem compared to 333 person-rem in 1990. The 1991 dose was the lowest in 22 years.[2]

Doses received during plant outages accounted for more than 87% of the annual collective dose for the 11 units with the highest doses in 1991. According to the report, the activities that most frequently contributed to these collective doses were valve maintenance and repair, in-service inspection work, control-rod-drive replacement and repair, installation and removal of scaffolding, and insulation and refueling activities. The report concluded that one way to reduce the annual collective dose of a plant is to reduce the frequency and duration of plant outages by detailed outage planning and scheduling of jobs to minimize critical path time.

INCREASED INSPECTIONS OF BWR STEEL CONTAINMENT STRUCTURES

The NRC recommended an increased inspection regime for 32 of the nation's nuclear reactors because of evidence that steel containment structures for some boiling-water reactors (BWRs) may be corroding more quickly than others. The reactors, all manufactured by General Electric, are those with either the Mark I or Mark II pressure-suppression containment designs. In the Nov. 20, 1992, edition of the *Federal Register*, the NRC solicited public comment on its inspection recommendations, which includes ultrasonic examination of the drywell and suppression-pool walls, and other efforts to detect leaks that could bring water in contact with the steel containment structures. In the notice, NRC said it was taking the action after GPU Nuclear discovered corrosion on the outside face of the steel drywell at its Oyster Creek plant. GE has been criticized in the past—and sued by several utilities—over contentions that the pressure-suppression design may not adequately condense steam in the event of an accident.[3]

RESTART OF TURKEY POINT 4

On Oct. 1, 1992, the NRC staff requested Florida Power and Light Company (FP&L) to suspend activities involving the restart of Unit 4 of the Turkey Point Plant in Dade County, Fla. J. Taylor, NRC executive director for operations, said FP&L agreed to shut down the plant following discussions between NRC and the Federal Emergency Management Agency (FEMA) about the status of emergency planning offsite. The plant was being returned to service following Hurricane Andrew and was at about 30% power when NRC requested the shutdown. In view of the damage done to the surrounding area by the hurricane, further consideration is being given by FEMA and NRC to the status of emergency planning in the area near Turkey Point. Neither Unit 3 nor Unit 4 at Turkey Point sustained damage to the nuclear portions of the plants from the hurricane, according to NRC.[4]

In late October 1992 FEMA approved the plant's plan for emergency evacuation in case of a nuclear accident and notified the NRC that it could approve a restart. The NRC immediately approved a restart and plant operators began start-up procedures. Unit 3 remained shut down until December because of refueling.[5] As a follow-up, in November 1992 the NRC staff and the Institute for Nuclear Power Operations (INPO) assembled a joint eight-member team that went to the Turkey Point nuclear power plant to compile experience gained from dealing with the damage resulting from Hurricane Andrew.[6] This team, together with FP&L, wished to gather a compilation of the lessons learned from the storm and its aftermath to benefit all nuclear utilities subject to hurricanes. The FP&L had already compiled much of this information during cleanup efforts to return the Turkey Point units to service. The joint team's study was to be separate and independent from normal NRC staff regulatory activities. The study was to cover the period beginning with FP&L's prior planning and preparations for the hurricane up to the time when off-site power was restored to the site. It was to focus on lessons learned related to facility operating and human resource decisions as a result of weather predictions and existing procedures and plans; emergency preparedness and emergency response as it related to utility internal actions; the impact on safety equipment from damage to non-safety-related equipment; damage to systems such as fire protection and plant lighting resulting from the hurricane and the respective compensatory measures taken; and human performance and needs related to shift staffing and coping, including issues associated with food and potable water supplies and site access.

ADDITIONAL TESTING TO RESOLVE PROBLEMS WITH USING THERMO-LAG FOR NUCLEAR PLANT FIRE PROTECTION

Despite concerns from some organizations [for example, the Nuclear Information and Resource Service (NIRS)], NRC spokesman R. Newlin stated at the end of 1992 that the Thermo-Lag 330-1 fire barrier system is "not an immediate problem." This sentiment was echoed in a statement released by the Commission which cited two specific examples for the low safety significance of the material. First, the compensatory fire watches adopted as a temporary measure have provided "an adequate level of fire protection." Second, multiple safety features, including automatic fire detection and sprinkler systems and the 24-hour-per-day availability of trained on-site fire brigades, make the ability of Thermo-Lag less essential.

One concern regarding Thermo-Lag is the possible presence of voids in the fire barrier, a gap in the material where a quick burn-through might be possible. Such voids were found at Comanche Peak Unit 2. Their possible existence was confirmed by R. Feldman of Thermal Systems, Inc. (TSI), makers of Thermo-Lag, to F. Miraglia of NRC.

Thermo-Lag was still being tested by both NRC and outside contractors. Several earlier fire tests gave varying results. Consistent with the Aug. 6, 1992, modification of the NRC's contract with the National Institute of Standards and Technology (NIST), a fire test was performed on a 1-hour gypsum board assembly in late September. The NIST had previously conducted six tests of Thermo-Lag material to evaluate the thermal/fire endurance performance of Thermo-Lag 1- and 3-hour panels. The NRC contracted with NIST to conduct these tests to assist in the closure of technical issues related to Thermo-Lag fire barriers. The NIST tested the gypsum board to assess the performance of this material against the performance of Thermo-Lag. The gypsum board assembly consisted of two $^{5}/_{8}$-inch sheets of fire-rated board separated by steel channels. The configuration was rated as a wall configuration, although it was tested in a horizontal position. The average thermocouple temperature on the unexposed side of the gypsum board assembly exceeded the temperature acceptance criterion of 325 °F in 53 minutes, whereas the 1-hour Thermo-Lag barriers exceeded this criterion in 22 minutes and 34 minutes, depending on the stress skin orientation and restraint. The staff was to consider the gypsum board test results as they related to results obtained from previous and future testing of Thermo-Lag.[7]

The fact that the gypsum board assembly lasted longer than the Thermo-Lag barriers was not surprising. However, the fact that the 1-hour rated gypsum board assembly did not last the full hour might be an indication that the small-scale NIST tests were more extreme than the standard tests used to establish barrier ratings.

Members of NRC's Plant Systems Branch witnessed Thermo-Lag 330-1 fire barrier fire endurance tests performed by Omega Point Laboratories (OPL), San Antonio, Tex., for Texas Utilities Electric Company on Nov. 3 and 5, 1992. The OPL and the licensee declared both tests successful on the basis of satisfactory post-fire inspections of the fire barriers and the cables. During the fire exposure, some temperatures measured on the conduit surfaces appeared to be inconsistent with visual observations. The temperatures were irregular and higher than expected. Post-fire inspection revealed that the insulation of some of the thermocouple wires located on the surfaces of the conduits were saturated with a substance that appeared to be a mixture of water and Thermo-Lag material. The OPL theorized that this affected the surface temperature measurements, which were therefore judged not to be useable.[8]

The first test assembly consisted of $^{3}/_{4}$-inch, 3-inch, and 5-inch conduit specimens with lateral bends and radius bends. The second assembly consisted of two 3-inch conduit specimens with junction boxes. The fire barriers included a combination of vendor-recommended assembly methods and licensee-designed upgrades. Both test assemblies were exposed to 1-hour ASTM E-119 standard fires followed by fog nozzle hose stream tests of 5-minute duration. Each cable's insulation resistance was checked by mega-ohmmeter before the fire exposure and immediately following the hose stream test. The OPL and the licensee determined that neither barrier was breached in any way. There were no burn-throughs, open seams, or open joints. Following the post-fire resistance tests, OPL removed the cables from the conduits and inspected each cable visually for fire damage. The laboratory and the licensee concluded that the cables were not damaged during the tests.

The results of another Thermo-Lag 330-1 fire-barrier fire endurance test, also performed by OPL, indicated that the assembly did not satisfy agreed-upon criteria for rated fire barriers.[9] The test assembly in this test consisted of $1^{1}/_{2}$-inch, $^{3}/_{4}$-inch, and 2-inch conduit specimens with lateral bends and radius bends. The fire barriers included a combination of vendor-recommended assembly methods and licensee designed upgrades. In addition to recording the maximum and average temperatures for the various samples, the licensee noted a burn-through on $1^{1}/_{2}$-inch and 2-inch conduit barrier material and stated that a post-fire Meg-Ohm-meter ("Megger") test was satisfactory, that there was some minor jacket swelling on the $1^{1}/_{2}$-inch conduit cable at two places and on one place of the 2-inch cable conduit. The conductor insulation was undamaged.

DISPLACED INSULATION IN SWEDEN

In October 1992 the NRC staff met with representatives of SKI, the Swedish Nuclear Safety Inspectorate, and STUK, the Finnish nuclear regulatory body, to discuss an event at the Barseback nuclear plant involving insulation blown off a pipe by a steam jet. The SKI staff displayed photographs and described the condition of the containment with the displaced pipe insulation. The pipe insulation was manufactured as a mineral-wool-type material and became displaced from its piping when a steam jet from a main steam relief valve prematurely discharged into the drywell. Following automatic actuation of the drywell spray system, approximately one-half of the displaced insulation was transported to the suppression pool where the debris clogged the containment spray suction strainers. The SKI had not yet determined whether either age or thermal degradation of the insulation had played a role in the insulation's dispersion into small particles with the appearance of a floc or fine fibers of wet paper.[10] The chief concern of the NRC is the fact that these dispersed materials pose a serious threat of plugging filters in a severe accident situation where water must be drawn from the containment sump of a reactor.

SOME NUCLEAR PLANTS MAY BE AT INCREASED RISK FROM MULTIPLE SG TUBE FAILURES

Some PWR nuclear plants equipped with Westinghouse steam generators may be at an elevated risk for a significant depletion of core coolant because of multiple SG tube failures.[11] NRC's Generic Issue (GI)

GI-163, "Multiple Steam Generator Tube Leakage," indicates that at Trojan the risk of a meltdown is 3×10^{-4}/yr, which is 300 times as great as the risk level proposed as acceptable by NRC's safety goal of 10^{-6}/yr.

A simultaneous rupture of as few as ten tubes could result in a release of radioactive water directly into the environment. The coolant water of the Emergency Core Cooling System (ECCS) could then also escape from the reactor containment through the broken tubes rather than being cycled back to the system to continue cooling the fuel core.

In the past the NRC had prohibited reactor operation if flaws or cracks in the steam generator tubes were deeper than 40% of the original tube wall thickness. However, in February 1992 Portland General Electric Company (PGE, the operator of the Trojan nuclear plant) had obtained a waiver allowing it to continue to operate, even though 428 tubes were found to be flawed. Several other Westinghouse plant owners also received license amendments allowing them to operate during one refueling cycle.

In response to the Trojan deferment, an NRC staffer filed a Differing Professional Opinion objecting to the waiver on the grounds that the risks to public safety may be too high. His concern applied to PWR operation with multiple steam generator tube through-wall cracks or other tube degradations. As a result, the NRC designated the problem of multiple steam generator tube leaks as a High Priority Generic Safety Issue.

In the GI-163 document, written by the NRC Office of Nuclear Regulatory Research (RES), it was pointed out that a PWR steam line break concurrent with steam generator tube rupture (SGTR) could result in a containment bypass loss-of-coolant accident. "If the SGTR involved enough tubes and if the main steam line break was not isolable, core damage would ensue upon refueling water storage tank depletion. The ruptured tubes and open steam line would then provide a direct path to the atmosphere for fission products from the deteriorating reactor core," according to the document.

The PGE disagreed with this assessment. According to PGE, the concerns have "little applicability" to the Trojan plant because they make assumptions that "don't apply." The PGE did specific strength testing and found that the flaws had "little or no effect on the strength of the tube." Nevertheless, the flaws described have been repaired.

Second, the concern regarding a main steam line break, "assume [that] the probability of ten tubes rupturing is a probability of one. Our [PGE's] analysis shows there is a one in 10,000 probability of a rupture."

A document distributed by the NRC Office of Nuclear Reactor Regulation (NRR) to RES disagreed with some of the research office's findings in the GI-163 document. The NRR report[12] said, in part:

> We believe the assumption of uniform thinning to be totally inconsistent with the actual degradation mechanism at Trojan. The assumption ignores the considerable body of evidence described in the staff's SER (Safety Evaluation Report) that the degradation mechanism at Trojan is dominantly axially oriented outer diameter stress corrosion cracking (ODSCC) with minor general intergranular attack (IGA) involvement. . . . There is no evidence of any uniform thinning at Trojan, either from the pulled tube data or from eddy current rotating pancake coil data. IGA involvement observed on pulled tube specimens was not sufficient to affect the burst strengths of the tubing; the observed burst strengths for the pulled tube specimens were found to be consistent with expected burst strengths for the limiting axial cracks based on measurements of the crack geometries performed subsequent to the burst tests. Even for tubes that exhibit significant IGA, the licensee for Trojan has demonstrated that such tubes would be expected to conform to the same burst pressure/bobbin voltage relationship as that developed from the pulled tube data and laboratory ODSCC specimens.
>
> In our opinion, the prioritization analysis does not provide a credible basis for concluding that implementation of the interim plugging limit at Trojan has created the potential for rupturing 10 or more tubes during a postulated steam line break (SLB) accident. This conclusion ignores the considerable evidence, discussed in the staff's SER, that all tubes can be expected to retain adequate integrity, consistent with the criteria of Regulatory Guide 1.121. This conclusion also fails to consider the probabilistic assessment of the potential for tube rupture during postulated SLB accidents that was included as part of the Trojan submittal.

SAN ONOFRE 1 PERMANENTLY SHUT DOWN

In early December 1992 Southern California Edison (SCE) shut down San Onofre Unit 1 (SONGS-1). It had generated electricity for nearly 25 years. The unit had begun commercial operation in January 1968 and was one of the nation's oldest nuclear power plants. The SCE officials decided to shut down the unit after determining that $125 million in needed repairs could not be economically justified. Unit 1, with an MDC rating of 465 MW(e), was the smallest of the three units at San Onofre and will not be replaced. The unit will be monitored by SCE until it is decommissioned, along with Units 2 and 3, in 2013 (Ref. 13).

Anticipating the closure, the NRC issued, in late October 1992, a contingent possession only license (POL) for Unit 1. This license amendment was to become effective once the facility was permanently shut down and the licensee had certified that all fuel has been removed from the reactor vessel and safely stored in the spent fuel pool. Defueling was to be completed by March 1993. When the POL amendment became effective, it removed the licensee's authority to operate the reactor. The POL establishes the basis for issuing various reliefs and exemptions from requirements that are not necessary to ensure safety in the permanently defueled mode. The licensee was required to submit a decommissioning plan for Unit 1 no later than 2 years after permanent cessation of operations.[14]

NINE NEW FINES DURING REPORTING PERIOD

Nine new penalty fines have been levied by the NRC on reactor licensees during the three-month period covered by this report (the fourth quarter of the year 1992). In each case the affected utility was required to report to the NRC on the causes and proposed corrections of the problem that led to the fine and had 30 days from the date of notification to either pay the penalty or protest its imposition in whole or in part. Each of the nine cases is briefly described here.

H. B. Robinson: Foreign Material in the RHR System

The NRC staff proposed a $50 000 civil penalty against Carolina Power and Light Company (CP&L) because tests at the H. B. Robinson Plant on Aug. 23–24, 1992, revealed that foreign material had been left inside the plant's Residual Heat Removal (RHR) system following a refueling outage between March 27 and June 24, 1992. A safety injection pump in the RHR system had to be declared inoperable on July 8, 1992, because of the presence of a plastic material, later identified as having come from plastic disks used during modifications on ECCS piping during the outage. The system was cleaned before being returned to service, but not all the plastic material was identified and removed. S. D. Ebneter, Administrator of the NRC's Region II, told company officials that the root causes of the violation were inadequate control of material and a failure to perform an adequate inspection following completion of the modification work. He said the use of plastic disks was not described by the modification work package and should have been approved before their use.[15]

Limerick: Failure to Perform Proper Radiation Surveys

The NRC staff has cited the Philadelphia Electric Company (PECO) for two violations of NRC radiation safety requirements at Limerick Unit 1 and proposed a $62 500 fine for the violations. One of the violations was reported to NRC inspectors by PECO for NRC review, and the NRC staff identified the other in the course of an inspection at Limerick on Sept. 22 and 29, 1992.

The NRC regulations require that radiation surveys be performed to evaluate any radiation hazards before a work group enters a radiation area and that workers be instructed on how to limit their exposure to radiation before beginning work. While making repairs on a valve in the containment building on July 7–9, 1992, several workers entered and worked in a high radiation area without adequate surveys being performed beforehand that would have identified an intense narrow beam of radiation passing through the work area, the NRC asserted, adding that failing to identify this beam created the potential for workers to receive exposures above regulatory limits, although this did not happen. In addition, some of the work crews moved from a surveyed area to an unsurveyed area because they had not been instructed to remain in the surveyed area. This led to the workers being exposed to the narrow beam of radiation as well as to general radiation fields at least six times as high as had been planned.

In a letter to PECO, T. T. Martin, Regional Administrator of NRC's Region I, wrote: "The fact that the individual exposures did not exceed regulatory limits does not diminish the NRC concern with the importance of implementing proper radiological controls over activities at the facility." The normal fine for such a violation is $50 000. In this case, NRC said, it was increased by 25%, to $62 500, because the licensee's self-assessment of the event failed to identify certain details, which demonstrated a lack of aggressiveness in determining the root cause of the event and identifying needed corrective actions.[16]

Wolf Creek: Reduced Heat Exchanger Water Flow

The NRC has proposed a fine of $50 000 against the Wolf Creek Nuclear Operating Corporation (WCNOC). The NRC took this action because, from July 21 to Aug. 27, 1992, there was a significant reduction in water flow through a heat exchanger that would have been needed had the plant experienced a loss-of-coolant accident during that time. This heat exchanger is used to cool plant safety-related equipment such as emergency cooling pumps during accident conditions. The NRC inspectors confirmed the condition after WCNOC discovered it and informed the agency.[17]

The reduced-flow condition was caused when a mechanic changed a valve position indicator while performing preventive maintenance on the valve on July 22, 1992. This action was beyond the scope of the job's written instructions and was not documented after it was done. As a result, the valve was left in the wrong position until WCNOC corrected the error on Aug. 27, 1992.

The WCNOC personnel determined that water flow through the heat exchanger was approximately 840 gallons per minute less than the normally expected amount of 8055 gallons per minute, which was about 80 gallons per minute less than the minimum flow specified in the plant's safety analysis report.

Once WCNOC had established that this nonconforming condition existed, it immediately repaired the valve involved and restored it to its required position. The company also checked to make certain that other similar valves were not mispositioned. It further developed plans to review pending preventive maintenance instructions in an effort to avoid such problems in the future.

The WCNOC analyses show that, despite the reduced water flow, plant safety equipment would have functioned under all accident conditions.

In his letter informing WCNOC of the penalty, James L. Milhoan, NRC regional administrator in Arlington, Tex., acknowledged those analyses. But he emphasized that the NRC is concerned because a violation of work controls "resulted in a significant reduction of the margin of safety." Milhoan added that this error "could easily have resulted in more significant degradation of essential service water flow because the mechanics involved did not recognize that they had affected flow by adjusting the valve position indicator incorrectly."

Limerick: Improper Firing of an Armed Guard

The NRC has proposed a $25 000 fine against PECO in connection with the firing of an armed guard by the guard's employer, a security contractor firm. A U.S. Department of Labor Administrative Law Judge (ALJ) upheld the former guard's claim that his firing was in response to his having voiced concerns about the Limerick site's security program. The contractor, Protection Technology, Inc. (PTI), appealed that decision to the U.S. Secretary of Labor.[18]

The NRC regulations forbid discrimination, harassment, or reprisal against an employee for raising safety concerns regrading nuclear power-plant operations regardless of whether or not the concerns ultimately are proven valid.

The NRC believed, on the basis of the ALJ decision, that, because the guard raised some concerns, his access to the Limerick site was denied, which resulted in his discharge by PTI.

In a letter to PECO informing the utility of this enforcement action, NRC Region I Administrator T. T. Martin referred to the finding by the ALJ and wrote, "shortly after the guard had engaged in protected activities, his supervisor retaliated against him because of that activity." The evidence showed, Martin said, that the primary motivating factor in PTI's decision to refer the employee for a psychological evaluation and, ultimately, to discharge him was his protected conduct in making a statement concerning Limerick security procedures. The ALJ based his decision, in part, on the fact that the employee was suspended the day after raising these safeguard concerns without explanation and without his having displayed any aberrant behavior. There was no documented evidence of prior behavioral/disciplinary problems with this employee. The NRC recognizes and fully supports, Martin went on to write, your need to pursue physical protection of your facility aggressively under the requirements of 10 CFR Part 73 and to assure fitness for duty for persons granted unescorted access to protected areas under 10 CFR Part 26. Nonetheless, you must also aggressively assure that individuals are not discriminated against for engaging in protected activities, as the ALJ found in this case.

Martin also said that the NRC staff proposed to fine PECO "primarily because of the actions of the person who was at that time the PTI site captain."

"Those actions are of particular concern because as the site captain, this person should have been responsible for protecting persons who raised safety concerns from harassment and intimidation. Such an environment cannot be tolerated if licensees are to fulfill their responsibility to protect the public health and safety. Thus, licensee management and licensee contractors must avoid actions that discriminate against individuals for raising safety concerns, and must promptly and effectively remedy actions that constitute discrimination," Martin said.

Indian Point 3: Inaccurate Information Given to NRC

The NRC has cited the New York Power Authority (NYPA) for three alleged violations of NRC requirements at Indian Point Unit 3 and has proposed a fine of $137 500 (Ref. 19).

One alleged violation is that NYPA officials gave the NRC staff inaccurate information during an April 10 conference regarding inoperable heating elements around pipes carrying boric acid solution; the heating is to prevent boric acid from crystallizing out of solution as it cools. Such crystals could obstruct flow and thus interfere with the operation of the boration system. A $100 000 fine was proposed for this violation.

The other two violations involve deficiencies in NYPA's Fitness for Duty Program. First, the utility allowed an NRC-licensed operator to regain unescorted access to the plant after he had tested positive for illegal drug use without confirming that illegal drugs had not been used after he had gone through NYPA's rehabilitation program. In addition, a second NRC-licensed operator apparently had not been periodically retested for illegal drugs after completing the rehabilitation program. The NRC proposed a $37 500 fine for these violations.

In a letter notifying NYPA of the proposed fine, T. T. Martin said, "...your staff presented information to the NRC, which was not accurate in all material respects, for consideration in determining the appropriate enforcement action associated with the issues discussed at the conference." Martin noted that NYPA made "no apparent attempt to deceive the NRC and [exhibited] no apparent willfulness with respect to the inaccurate presentations made at the conference."

In regard to the other two violations, Martin wrote that the reassignment of an operator after failing a urine test without adhering to policy was of "significant concern to the NRC" and that the lack of a follow-up test for the second violator was of "particular concern to the NRC since the lack of a follow-up testing program was identified during the initial investigation of your Fitness for Duty program" on Dec. 9, 1991.

Point Beach 2: Plugged Piping Makes a Safety System Inoperable

The NRC has proposed a $75 000 fine against Wisconsin Electric Power Company (WEPCo) for operating Point Beach 2 while a portion of a safety system was inoperable because of a temporary plug left in the piping following a modification in 1991. The material was found in a pump that is part of the plant's emergency systems. A test on Sept. 17–18, 1992, showed abnormally low water flow through the pump; the utility dismantled the pump and found a foam disk blocking the pump internals. The disk had been left in the piping following a system modification during the fall 1991 refueling outage. The disk had been used to keep foreign materials out of the piping during the modification.

The affected pump and piping system would be called upon to function under some configurations of the emergency core cooling system in the event of a major accident. A parallel pump and piping system would have been available to perform the safety functions of the degraded system.[20]

The company was cited also for failing to have an adequate procedure to ensure that the piping was free of obstructions following the system modification in 1991.

As corrective action, the utility thoroughly inspected the safety-system piping of both Unit 1 and Unit 2 to ensure that there were no additional obstructions. Plant procedures were revised to control modification work better and to identify and remove any construction-related debris.

WNP 2: Core Power Oscillations

The NRC staff is proposing to fine the Washington Public Power Supply System (WPPSS) $75 000 for violations of NRC requirements regarding the maintenance of reactor core stability. These violations were related to an Aug. 15, 1992, power oscillation event at Washington Nuclear Project No. 2 (WNP-2).[21]

During an inspection conducted Oct. 5–21, 1992, NRC inspectors identified the following alleged violations: (1) use of procedures that did not provide adequate instructions or acceptance criteria for developing control-rod patterns that would prevent core power oscillations; (2) failure of the WNP-2 Nuclear Safety Assurance Group to review a Boiling Water Reactor Owners' Group letter regarding potential reactor power oscillations; and (3) inadequate review of the mixed core designs used for the summer 1992 reload and the previous reload of the WNP-2 reactor core.

The base penalty for this violation is $50 000. However, the base penalty was increased because of poor past performance in management and quality oversight and because industry and NRC guidance regarding reactor power oscillations was available to WPPSS staff.

River Bend: Radiation Safety Errors

The NRC has proposed a fine of $100 000 against Gulf States Utilities (GSU) for violations of NRC radiation protection requirements at the River Bend Station. This enforcement action was based on findings from inspections conducted in response to two incidents that occurred in the fall of 1992 while River Bend was shut down for refueling.[22] The first involved the mislabeling of a plastic bag containing low-level radioactive waste. A tag on this bag indicated that radiation levels on contact were less than 2 millirem per hour, whereas surveys showed the levels actually to be as high as 14 000 millirem per hour. The NRC found no indication of an overexposure in this case.

The second incident was the release from the plant site of a piece of scrap steel slightly contaminated with radioactive material. It had been sent to a Baton Rouge, La., scrap iron facility where it was discovered on Oct. 19, 1992. The GSU confirmed that the item had originated at River Bend and brought it back to the plant.

As the result of an inspection scheduled in response to these events, the NRC cited GSU for 16 violations of NRC radiation safety requirements, many of them related to the scrap steel and mislabeled bag incidents. A number of them were discovered by GSU during its own investigation. NRC's enforcement action was based on eight instances of failure to perform required radiation surveys, five instances of failure to follow radiation protection procedures specified in the plant license, two instances of a failure to supply radiation monitoring equipment to workers in appropriate circumstances, and one instance of radioactively contaminated material being transferred to an unauthorized recipient.

In his letter informing GSU of the proposed fine, NRC Regional Administrator J. L. Milhoan took note of both short-term and long-term corrective measures GSU either has taken or had under way to eliminate weaknesses in its radiation protection program. He acknowledged that it did not appear that the violations resulted in overexposures to individual workers.

Nevertheless, Milhoan added, the situation posed the potential for exceeding exposure limits because, he said, the violations "represent a breakdown in the controls that are essential to the prevention of significant radiation exposures and to the control of radioactive material."

Oconee 3: Reduced Flow in Service Water System

The NRC staff proposed a \$100 000 penalty against Duke Power Company for an alleged violation of requirements at the Oconee Nuclear Power Plant, Unit 3. In the Notice of Violation sent to the company, NRC officials said the proposed action had been taken because company employees identified the existence of a reduced flow condition in the low-pressure service water system of Oconee 3 but had failed to take adequate corrective action to restore the flow rate to its proper level. The NRC viewed it to be a significant safety issue that the plant operated from June 9, 1992, until Sept. 14, 1992, with the inadequate cooling flow.[23]

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15. NRC Press Release 92-156 (Oct. 26, 1992).
16. NRC Press Release 72-170 (Nov. 27, 1992).
17. NRC Press Release 72-177 (Dec. 8, 1992).
18. NRC Press Release 92-180 (Dec. 9, 1992).
19. NRC Press Release 92-181 (Dec. 10, 1992).
20. NRC Press Release 92-182 (Dec. 10, 1992).
21. NRC Press Release 92-186 (Dec. 16, 1992).
22. NRC Press Release 92-190 (Dec. 29, 1992).
23. NRC Press Release 92-191 (Dec. 29, 1992).

Recent Developments

Edited by E. G. Silver

General Administrative Activities

Compiled by M. D. Muhlheim and E. G. Silver[a]

"General Administrative Activities" summarizes selected current topics related to nuclear safety that do not fit elsewhere in the journal. Included in this issue are items reported during October, November, and December 1992. Subjects discussed, among others, are Advisory Committee on Reactor Safeguards (ACRS) comments on digital instrumentation and control systems, radiation effects from Chernobyl fallout, former Soviet nuclear reactor dispersals on the ocean floor, a medical radiation source mishap with serious consequences in Pennsylvania, and an update on Cuban plans to build two nuclear power plants.

ACRS COMMENTS ON SEVERAL ISSUES

The ACRS issued a number of letter reports to the Nuclear Regulatory Commission (NRC) during the period covered by this report (October, November, and December 1992), several of which will be discussed and excerpted here.

Digital Instrumentation and Control System Reliability

During its Sept. 10–23, 1992, meeting, the ACRS reviewed the NRC staff's proposed approach to defense against common-mode failure of digital instrumentation and control (I&C) systems, as discussed in the draft Commission paper "Design Certification and Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light Water Reactor Designs," forwarded to the Commission on June 25, 1992. Specific comments on policy issue "A" of this document are reported in a letter to Mr. Taylor dated Sept. 16, 1992. The concerns raised are more generally applicable, however, for example, to the staff's proposed generic letter on digital replacements for analog systems. Their report reads, in part:[1]

> The trend in most industries over the last few decades has been toward the replacement of analog instrumentation and control systems with digital alternatives, and the nuclear industry has been no exception. This has been true for both functional replacements within existing nuclear facilities and for new designs, so it has been necessary for the staff to develop regulatory practices to deal with both the novel opportunities and the novel threats posed by these systems.
>
> Experience, both military and industrial, has generally shown the digital systems to be more reliable and versatile than their analog counterparts. There are, however, some caveats and some regulatory conundrums. An advantage is that the digital systems are capable of more complex functions, so it is possible to build in self-testing capabilities that provide continuous assurance of operability with negligible system stress. In addition, the digital systems don't wear out; a billion activations of a CMOS gate are no more damaging than a thousand. While much has been made of the vulnerabilities of multiplexed data transmission systems, some of which are doubtless real, such systems generally provide greater fidelity and reliability of data transfer, along with greater fault tolerance through error-correcting coding. (If an analog signal is corrupted, it is often not possible to know it has happened.) Indeed, error detection and error correction can be carried to arbitrary lengths for digitized data. There are many other advantages, and the future clearly belongs to digital systems, where they can be used.
>
> On the negative side, the available complexity of function afforded by digital systems invites the creation of complex software, which can be difficult to validate and can be

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subject to surprising error modes. Such systems are also hard to regulate, because only the simplest programs are amenable to formal validation and verification, in the sense of a complete analysis of the mapping of the input space to the output space. For more complex programs (relevant to nuclear control systems, but not necessarily to instrumentation or safety actuation systems), there are many analytical techniques in use, none perfect. That is also true of analog systems. Solid-state systems, whether digital or analog, are also peculiarly vulnerable to environmental damage, e.g. from overheating. Finally, programmable digital systems have their own special vulnerabilities to human error.

The staff has concentrated its attention on one of these many issues, the vulnerability of digital systems to certain kinds of common-mode failures, principally through programming errors introduced into the software, and therefore common to all channels.

To deal with this supposedly special susceptibility to common-mode failure, the staff has proposed a set of regulatory requirements. The set includes some unarguable items, like the provision of adequate diversity to cope with common-mode failures that can affect safety systems, and analysis of the appropriate accident sequences. The set also includes some items whose desirability is less clear, and we now turn to these . . . [in] no special order.

The lack of explicit and quantifiable safety standards for instrumentation and control systems is particularly troublesome here. The staff speaks of reliability for digital systems in the same terms (failures per demand) that it uses for items which do wear out, like relays and switches. The entirely different failure mechanisms make this an inappropriate transfer of terminology. Indeed, a simple software-based system, in which the hardware is kept within its environmental constraints, and whose software is simple enough to have been subjected to a full validation and verification (in the sense used above) can be expected to never fail. (Never is only a slight exaggeration.). The failure anecdotes we all know are typically in systems that are too complex for formal V&V, leaving the door open to software errors, or have been mistreated, opening the door to hardware failures. The latter problem is not unique to digital systems.

In view of the lack of explicit standards for the reliability of the digital systems, the staff seems to have drifted to what has been called the "bring me a rock" posture, in which the industry is asked to analyze its own vulnerabilities, after which the staff will make its ruling about the adequacy of the design. The spirit of the safety-goal initiative was presumably to help make regulation more predictable, and this approach is clearly in the other direction.

The focus on common-mode failures is troublesome. Software errors in single systems can lead to accidents just as serious as those due to common-mode failures in redundant systems, and the entire question of software reliability greatly transcends the issues raised here. We have been conducting a coordinated series of meetings on the safety issues involved in the inevitable computerization of the industry, already in progress. When we report on these, we will doubtless raise the question of whether sufficient talent, both in quantity and in experience, is being directed at these issues by NRC. That question is also an underlying issue here.

For the specific issue of protection against common-mode failures, whether for digital systems or such devices as diesel generators, there is a set of standard prophylaxes like diversity and defense in depth, which are useful when applied sensibly. (Slogans can be overplayed. It makes no sense to insist that multi-engine aircraft have a suitable mix of turbine and piston engines.)

The most controversial specific position taken by the staff is that there must be a safety-grade set of displays and controls located in the control room, independent of the computer systems, and "conventionally hardwired" to the lowest level practicable. Though the intent of the words in quotations is unclear, we were assured that it was to require analog backup systems. We do not concur in this proposed requirement. We think that the staff is unnecessarily mixing up the issues of digital/analog, hard wire/multiplex, and software/hardware.

Each instrumentation and control system that is important to the safety of a plant ought to meet some identifiable standard of reliability and fault tolerance, regardless of the hardware/software basis used in designing and fabricating the system. It is not necessary that any given element of the system be perfect, but that the system as a whole meet some recognized standard, presumably in the form of a relevant surrogate for the Commission's safety goals. Both the identification of that standard and the evaluation of conformance for the system in question pose problems, but each should somehow be completed before, not after, a regulatory position is established. For example, the staff proposes to require that a backup system provide protection equivalent to that of the primary system, whereas the need is for sufficient protection to assure the adequate safety of the plant. It is not at all uncommon for backup systems to be designed to lower standards than the primaries, taking into account the fact that they will be called upon less often. (Consider spare tires.)

It is entirely possible that a digital system may turn out to be a better backup than an analog system. (The proposed position does accommodate this idea, but the staff briefings did not.) For some situations a light beam is a more reliable means of communication than a hard wire. A general-purpose microprocessor that is in widespread commercial use may be more reliable (and more thoroughly tested) than a special-purpose analog switch. And so forth.

In each case it is necessary to make a specific reliability analysis, measured against a reasonable standard, and the staff gave no evidence of having done so for any case. Instead, it has adopted a general requirement for an analog backup for all cases, and we were not convinced by the justification provided.

We recommend that the staff revisit these issues, augment its own capabilities, and broaden its interaction with those elements of the outside world who have previously dealt with such problems. It would be unwise, however, to read too literally into the nuclear arena the considerations that are relevant to far more complex systems. We are dealing here with the relatively simple safety-centered parts of the computerized instrumentation and control system, and an architecture that exploits this fact may be more robust.

Comments on the Draft Commission Paper, "Design Certification and Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light Water Reactor Designs"

The report reads, in part:[2]

Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems. It is our view that the thrust of the staff recommendations concerning defense against common-mode failures in digital I&C systems as underlined in Issue A of the draft Commission paper is appropriate. We agree with the staff that the applicant should be required to assess the defense in depth and diversity of the proposed designs for the events postulated in the Safety Analysis Report, and demonstrate an acceptable plant response for each. The staff proposes that the instruments, controls, and equipment required to demonstrate an acceptable response be independent of any common-mode failure mechanisms associated with the event. We view this requirement to be essential, but remain open as to the best approach. The staff proposes an independent set of safety-grade displays and controls in the main control room. We believe that other arrangements might be shown to be acceptable.

Analyses of External Events Beyond the Design Basis. To assist in the closure of severe accident issues, the staff recommends that (1) analyses submitted in accordance with the requirements of 10 CFR 52.47 (concerning the contents of applications for standard design certification) include an assessment of internal and external events and (2) during the design certification review, the staff should evaluate those external events that are not site dependent (e.g., fires, internal floods) and certain bounding analyses. We agree with this staff recommendation.

Elimination of the Operating Basis Earthquake from Seismic Design. The staff is still reviewing this issue and has expressed only an interim position. We believe the staff is taking an appropriate approach in its interim position.

Multiple Steam Generator Tube Ruptures. The staff is recommending that the applicant for design certification perform additional analyses to determine the AP600 response to multiple breaks of up to five steam generator tubes. We agree with the staff's recommendation, but believe the staff should have a better technical basis for estimating the frequency of occurrence of such multi-tube breaks.

The staff is also recommending that the applicant for design certification of a passive or evolutionary PWR assess design features necessary to mitigate the amount of containment bypass leakage that could result from MSGTRs. We agree with the staff's recommendation.

Probabilistic Risk Assessment Beyond Design Certification. The staff is recommending that, throughout the duration of the combined or operating license, the PRA be revised to address significant plant modifications, operating experience, and other developments that may affect previous PRA insights. We are convinced that it is worthwhile for a plant operator to have an up-to-date PRA and are, therefore, reluctant to recommend against this position. However, if this is to be required, the staff should more clearly specify how it intends to use the updated PRA and what is meant by keeping it current. We think such guidance is part of the overall issue of appropriate use of PRAs in regulation and would be helpful to licensees and to the staff.

Role of the Operator in a Passive Plant Control Room. We agree with the first part of the staff's position "that sufficient man-in-the-loop testing and evaluation be performed . . . to demonstrate that functions and tasks are integrated properly into the man/machine interface design" of passive ALWR control rooms. . . .

Control Room Annunciator (Alarm) Reliability. We agree with the staff's position that the alarm system for ALWRs should meet the requirements of the EPRI Utility Requirements Document.

Regulatory Treatment of Nonsafety Systems. We were told that the staff is still engaged in significant ongoing discussions and review of this issue and that the associated position and recommendations are subject to modification. We believe the issue is substantial and has broad implications with respect to such items as use of PRAs in regulation, safety goal implementation, and reduction of regulatory burdens, and we expect to have additional future interactions with the staff and the industry. Consequently, we are not prepared to express a position on this issue at this time.

GE Nuclear Energy's BWR Power Uprate Program

The ACRS reviewed the General Electric Nuclear Energy generic program supporting power uprates for operating boiling-water reactors (BWRs) and the associated application by Detroit Edison (DECo) for a power level increase for Fermi 2. Their report reads, in part:[3]

DECo has requested an amendment to its technical specifications to increase the licensed thermal power limit from 3293 MW(t) to 3430 MW(t), a 4.2 percent increase . . . based on the generic BWR power uprate program developed by GE. For this program, the staff has limited the core power increase to no more than five percent. Licensees for twenty BWR units have expressed interest in similar power up-rates pursuant to this generic program. The DECo uprate request represents the lead plant effort.

Nine U.S. BWR units are licensed to operate at the up-rated power and, as a result, there are 229 reactor-years of operational experience. Many BWRs have the capability to increase core power well beyond the five percent limit assigned to the GE generic uprate program at this time.

Power increases of 15-20 percent have already been accomplished at BWR nuclear power plants located overseas, albeit at some additional hardware expense. The Fermi plant will still have at least an additional five to ten percent margin in its safety systems (using their design basis) following adoption of this uprate.

We concur with the staff's conclusion that there is reasonable assurance that the health and safety of the public will not be endangered by the proposed power up-rates, and that DECo should be issued its requested amendment. We commend the staff, DECo, and GE for a job well done. The detail in the staff's analysis represents a thorough safety evaluation and clearly supports its conclusions. We do, however, offer the following comments for consideration. . . .

During this review, it came to our attention that the design basis for plant equipment is used in analyses supporting determination of safety margins. This is done in spite of demonstrated substantial equipment performance margins. This is an example of unnecessarily compounded conservatism. Safety margins should be determined using actual data, when available.

Proposed Guidance for Implementation of the Maintenance Rule, 10 CFR 50.65

On Oct. 8–10, 1992, The ACRS reviewed the NRC staff's proposed guidance regarding implementation of the maintenance rule, 10 CFR 50.65. This rule is to become effective on July 10, 1996. The report reads, in part:[5]

The package of documents, which consists of a proposed regulatory guide and other supporting documentation, describes the staff proposal to endorse an industry consensus guidance document (Draft NUMARC 93-01) to implement the maintenance rule. The industry has a demonstration program in progress involving implementation of this guidance at nine nuclear power plants. The staff points out that its endorsement of this document maximizes "the leadership role of the industry in the area of maintenance." The staff believes that, "The performance based, results oriented characteristics of the maintenance rule make industry cooperation vital to successful implementation of the rule."

We agree with the staff's position and recommend that this package be issued for public comment.

We plan to review the staff's proposed final implementation guidance for the maintenance rule after the staff has resolved public comments, and to provide our comments to the Commission.

As presently proposed, the scope of the monitoring program with regard to the electrical connections to the utility transmission network is unclear. We recommend that the staff's final guidance be extended to include the switchyards.

During our meeting, we asked the staff to describe the progress it had made on developing guidance to the industry for implementing a maintenance program to satisfy the maintenance rule, and which also addresses the requirements of the license renewal rule. . . . Based on our discussions with the staff, we believe that continuing senior staff management attention to this issue is needed in the interest of coherence in the regulatory process. We also note that the reliability assurance programs being required of ALWR licensees will involve the establishment of a third kind of maintenance program. Consistent staff guidance is needed on the elements of an acceptable program that will satisfy these three sets of requirements.

Proposed Technical Position on Environmental Qualification of Electrical Equipment for License Renewal

On Oct. 8–10, 1992, the ACRS also reviewed a proposed Branch Technical Position (BTP) on Environmental Qualification of Electrical Equipment for License Renewal. Their report reads, in part:[5]

Under the License Renewal Rule, 10 CFR Part 54, applicants will be required to develop a comprehensive program to identify in their plants all structures, systems, and components which may be subject to age-related degradation unique to the license renewal period. A further program to manage these components to ensure continued safe operation of the plant is also required. The staff is now proposing an additional program, by means of a BTP, which singles out environmental qualification of electrical equipment for special treatment in the license renewal period. The particular concern of the staff seems to be that the qualification standards for insulation used on electrical cables prior to 1984 (representing 87 of 111 licensed nuclear power plant units) may not ensure adequate performance of cables for extended plant life. That, of course, is the issue for all SSCs in a plant, and it is not clear to us why the more general treatment of SSCs called for under 10 CFR Part 54 is not adequate for electrical cables as well.

Industry representatives expressed objection to the staff proposal for a BTP. They believe that while older plant cables were qualified to a lesser standard than has been in use since 1984, these cables have been approved for continued use in the plants (as has much other equipment where standards have evolved) and are part of the Current Licensing Basis for each of these plants. Their interpretation of 10 CFR Part 54 is that the CLB is to be preserved with the exception that those SSCs subject to age-related degradation unique to the license renewal period should be subjected to specific management programs. They see no need for the BTP and believe it will result in unnecessary cable replacements and add significantly to plant costs for license renewal.

We are not convinced that the proposed BTP has been shown to be necessary or appropriate. It should not be issued for public comment until the matters discussed below have been addressed.

Neither the staff nor the industry presented any risk perspective on this issue. In simple terms, the risk is as follows: During the license renewal period the electrical cable in a key system might degrade in a way that the degradation would remain undetected during normal operation and by normal maintenance, testing, and surveillance practices. Then, during an accident, i.e., a LOCA, the insulation would fail and the key system would not perform its design function to mitigate effects of the accident. Present licensing practice assumes, and experience seems to confirm, that the probability of this sequence during the initial license period is acceptably low. At issue is whether the probability during the license renewal period is significantly greater. No evidence has been presented either way. Analysis of the risk importance of this issue should be made before the BTP is finally accepted or rejected. Such an analysis should include estimates of downside risks inherent in major projects intended to improve nuclear power plant safety.

Many electrical cables are covered with fire retardant materials. These coatings could have important effects on the aging of the cable insulation. Apparently, these effects have not been considered by the staff in development of this BTP. We do not know whether they have yet been explicitly considered in the selection and evaluation of important SSCs in license renewal programs. They should be.

Proposed Amendments to 10 CFR Part 55 on Renewal of Nuclear Power Plant Operator Licenses and Requalification

At the same Oct. 8–10, 1992, meeting, the ACRS reviewed proposed amendments to 10 CFR Part 55. It reported, in part, as follows:[6]

> These proposed amendments would revise the current requalification regulations for licensed operators at nuclear power plants by eliminating the present requirements that they pass a requalification written examination and operating test administered by the NRC during their six-year license term. Licensed operators would continue to be required to pass the biennial requalification written examination and annual operating test administered by their plant training organizations. As part of the proposed rule change, licensees would be required to submit their examinations and operating tests for NRC review. The staff points out that these changes in the regulations will allow the redirection of NRC license examiner resources so that the examiners will be able to perform more comprehensive, programmatic inspections of licensee operator training programs.
>
> We believe that these proposed amendments to 10 CFR Part 55 will be beneficial and recommend that they be released for public comment. We would like the opportunity to review the proposed final version of these amendments after the staff has reconciled the public comments.

Environmental Qualification for Digital Instrumentation and Control Systems

During its Oct. 8–10, 1992, meeting, the ACRS reported on environmental qualification (EQ) for digital I&C systems. Its report reads, in part:[7]

> As part of its continuing effort to meet the challenges posed by the emergence of modern digital instrumentation and control systems, the staff is concerned about the peculiar vulnerabilities of such systems to environmental stress. There is, therefore, a research program ... directed at uncovering enough information to provide regulatory guidance. The program is far from complete. We were told that it will ultimately study about a dozen environmental stressors, including temperature, moisture, smoke, etc., but the preliminary results presented to us were in fact confined to the area of EMI/RFI (electromagnetic/radio-frequency interference).
>
> We were told that the staff had made no effort to set priorities or to assess the risk levels associated with the various stressors before deciding to concentrate on EMI/RFI, and are therefore concerned that it may be emphasizing the problem easiest to solve, rather than the most risk-significant. A coherent approach to risk management and regulation would assign the NRC's scarce resources and expertise through risk-based criteria.
>
> Our judgment (in fairness, also not based on detailed priority analyses) is that the problems of EMI/RFI are receiving unwarranted emphasis. This is not to say that they are unreal—there are many anecdotes of interference-induced failure—but only that the nature of the threat and of its solutions are well understood, from work done in different contexts. Careful attention to shielding and to grounding, together with electromagnetic discipline when shielding is compromised (as, perhaps, by opening metal cabinets), can go a long way toward alleviating any vulnerabilities that may exist. The techniques are well known, and in no way mysterious.
>
> Indeed, in the military world, where susceptibility to intentional jamming is a constant threat, and even vulnerability to extremes of temperature, moisture, and smoke is an endemic concern, there is an enormous body of information about measures and countermeasures. We were therefore surprised to be told that NRC had made no contact with the relevant agencies before embarking on its own research program.
>
> We do agree that the NRC must develop guidance for the protection of vital electronic systems (and indeed for all other vital systems) from potentially disabling environmental influences, but we heard no rationale for the specific concentration on the one threat singled out for attention.
>
> We recommend that the direction of the program be reassessed to account for some kind of risk ordering of a suite of likely stressors, and that diligent efforts be made to draw on the experience of the community, including the military community, for relevant information. None of these phenomena are unique to the nuclear world.

Risk-Based Regulation

During a Nov. 5–7, 1992, meeting, the ACRS reviewed a draft Commission paper on Risk-Based

Regulation. The paper responds to a Staff Requirements Memorandum dated Mar. 26, 1992. Their report reads, in part:[8]

> We interpret the Commission's charge to the staff as reflecting a recognition of the increasingly sophisticated and widespread use of analytical risk assessment techniques in the nuclear enterprise, a natural evolution of a process that began with the 1975 publication of the Reactor Safety Study, WASH-1400. Since it is now possible to make informed and quantitative statements about many (but not all) of the contributors to nuclear risk, it is correspondingly possible to optimize the deployment and use of the regulatory resources available to the Commission. The SRM directed the staff to both examine the feasibility of such a risk-based approach to regulation and to suggest means by which it could be implemented. The draft paper on which we were briefed is the preliminary response to that charge.... [I]mportant to us is the issue of coherence of the various efforts now in progress in various parts of the staff to develop and implement activities that could be collected under the name of risk-based regulation. We have commented earlier about the Maintenance Rule, Regulations Marginal to Safety, and other initiatives involving the use of risk analysis, and have at this meeting heard about Risk-Based Regulation, revision of the Regulatory Analysis Guidelines, and the Prioritization of Generic Safety Issues. Each of these requires informed use of quantitative risk information and appropriate attention to the Commission's safety goals, yet each is being analyzed by an independent group, with an independent perspective on the NRC's needs. In addition to this, there is the PRA Working Group, whose progress we have been following closely. We are unable to find any focal point for all these efforts, except at the level of the EDO.
>
> We continue to call for increased coherence in the treatment of all these matters, bound to each other by the common need to weave the threads of the safety goals (the expression of the ultimate objective of regulation) and quantitative risk assessment (the tool that makes more directed risk management possible) into the NRC fabric. If it is not done at the level of the EDO it will not be done, and resources that could be devoted to assuring nuclear safety will be squandered.
>
> In the past we have suggested strong measures to address this problem. While not pushing any particular solution, we still believe that the collection of issues discussed here is important to the future performance of the agency. The coherence problems will not be solved by an incoherent effort.

JAPANESE NUCLEAR POWER PLANT TRIPS REACTOR COOLANT PUMPS

A mistaken flip of a switch by a nuclear power plant worker caused a Japanese reactor's cooling pumps to trip, but emergency systems responded by shutting down the plant core and flooding the core with water.[9] No radiation was released in the accident at the Fukushima Nuclear Plant, 70 miles northeast of Tokyo. The plant is a 784-MW(e) pressurized-water reactor.

Local officials sharply criticized the Tokyo Electric Power Co., the plant's owner, for not notifying residents for hours about the emergency shutdown of the 18-year-old reactor. "The case is very serious because it triggered the emergency core cooling systems," said J. Takagi, a physicist who heads Japan's Citizens' Nuclear Information Center.

The accident was caused when a plant operator mistakenly flipped a switch and thus caused a computer to believe that a backup water pump was operating when it was not, said R. Fujii, chief of the Ministry of International Trade and Industry's nuclear safety division. The computer then automatically shut off the reactor coolant pumps. The insufficient supply of water to cool the reactor caused a group of backup pumps to fail, Fujii said. He said plant operators corrected their mistake within a minute, but the water level already had dropped dramatically. Emergency systems shut down the plant and provided the necessary heat removal capacity.

GAO RELEASES REPORT ON DOE TEST AND RESEARCH REACTORS

In response to concerns expressed by a member of Congress, the General Accounting Office (GAO) released a report asserting that the DOE must soon decide on which research reactors it will shut down, maintain, or replace to avoid possible degradation in safe operation, increased operating costs, degradation in performance, and gaps in needed reactor service.[10]

In February 1991, Rep. M. Synar (D-OK), chairman of the House Environmental, Energy, and Natural Resources Subcommittee, requested that GAO examine a number of issues regarding the Department of Energy's (DOE's) nondefense-related nuclear facilities. Synar was particularly concerned that test and research reactors are aging and may eventually become unsafe to operate.

In April 1992, DOE was operating 10 nondefense test and research reactors, down from over 25 at the beginning of the 1980s. Both the program and the need for a large number of reactors to support it have diminished, however, mainly because of a decrease in demand for reactor services, according to the GAO report. The decrease in the demand has been accompanied by an increase in operating costs resulting largely from expanded safety requirements and standards. With demand for the services of some of its reactors decreasing, the DOE decided that it would not be cost-effective to continue to operate all these reactors, the GAO found.

The DOE ensures safe operation of reactors through continual inspections performed by staff from DOE headquarters and field offices as well as by the contractors responsible for the operation of the reactor facilities. The DOE has required Safety Analysis Reports (SARs) for all its nuclear facilities since 1976, according to GAO. The GAO Report said, however, that DOE has not developed a long-range plan for use of its nondefense test and research reactors. In DOE's view, the diminishing number of these reactors and their individual uniqueness make a formal plan to manage most aspects of their use unnecessary. The DOE did concede, however, that it would eventually have to make decisions and plans concerning the retirement or replacement of its aging test and research reactors. The agency began to develop a long-range use plan for its test and research reactors in 1987, but it remained in draft form and has not been worked on since 1987, GAO found. The DOE believes that the number of reactors and the need for some of them had changed or diminished so rapidly that it was difficult and perhaps unrealistic to develop a strategic plan for managing the reactors.

The National Research Council and the Advisory Committee on Nuclear Facility Safety are especially concerned that DOE needs to prepare plans for the eventual retirement or replacement of aging research reactors, the GAO reported. The groups contend that if DOE sees that long-term reactor missions will need to be supported, it should move ahead expeditiously to plan for the replacement of aging reactors. The Council and the Committee both believe that timely planning and execution of reactor retirement and replacement projects "can alleviate potential safety concerns about the operation of aging reactors and preclude gaps in reactor availability such as those that occurred recently with DOE's production reactors."

In addition, the Committee has suggested that DOE should accelerate its planning for the replacement of its basic research flux reactors, such as the High Flux Isotope Reactor. The Committee pointed out to GAO that problems resulting from aging have prompted special surveillance of irradiation damage to materials at these reactors and that, as a result of this continuing damage, these reactors may be available only for a limited additional time.

RADIATION EFFECTS FROM CHERNOBYL FALLOUT

Medical tests on people in Belarus affected by the accident at Chernobyl in 1986 have brought "alarming results," A. Volkov, director of the Independent International Radiation and Environment Centre and a member of the Belarus parliamentary commission on Chernobyl, told a German press agency in November 1992. Volkov said that during the first 5 years after the catastrophe, about ten times as many children as normally expected became ill with cancer of the thyroid gland. There were 27 cases of such cancer in 1990, 55 cases the following year, and 30 cases in the first half of 1992. Nearly all children being treated at the Minsk Research Institute for Radiation Medicine showed a disturbed hormone system, Volkov said. The number of related diseases and psychic disturbances had sharply risen in the irradiated regions of Gomel and Mogilyov as well as in parts of the Minsk and Brest regions, according to Volkov.[11]

RUSSIANS DISCLOSE SUBMARINE BURIAL GROUND

Russian authorities revealed in late 1992 the exact locations where four nuclear submarines containing missiles and torpedoes have sunk. The Russians released the information in a meeting with an American delegation at the Woods Hole Oceanographic Institution in Massachusetts. They also disclosed the area, near the Arctic island of Novaya Zemlya, where several Russian reactors and other radioactive waste were dumped over the last 30 years. Besides four submarines lost at sea, the Russians asserted that several decommissioned Soviet Navy nuclear reactors were dumped in a shallow part of the eastern Arctic Ocean. In addition, three reactors from the nuclear-powered icebreaker *Lenin* were dropped into the Sivolky Gulf in 1967. In 1972, a barge with a submarine reactor was sunk in the Kara Sea. In 1982, the submarine K-27 was jettisoned after an emergency with two fuel-laden reactors in Stepovov Gulf. Then, in 1988, a reactor was dumped in Techeniya Gulf. But perhaps of greater concern is the radioactive waste dumped at sea. Russian authorities told the delegation that 11 000 to 17 000 waste containers, containing 61 407 curies of radioactivity, were dumped off Novaya Zemlya from 1964 to 1990. In addition, 165 000 cubic meters of liquid waste were dumped in the Barents Sea west of Novaya Zemlya from 1961 to 1990. For comparison, the Chernobyl accident released about 86 000 000 curies of radioactivity.[12]

LOSS AND RECOVERY OF A MEDICAL TREATMENT RADIATION SOURCE

The NRC staff investigated the loss of a radioactive source from a medical treatment facility in Indiana, Pa., on Nov. 16, 1992, and its subsequent discovery when it set off an alarm on a radiation monitor at the entrance to a waste disposal facility in Warren, Ohio, on November 27, 9 days later.[13]

According to information received by NRC, the source had been used on November 16 in the Indiana Regional Cancer Center of Oncology Services Corporation to treat a cancer patient. Officials of the facility said they performed no radiation treatments with the source after that date.

The NRC reported that the source had been recovered and returned to the medical facility. It is a tiny piece of iridium-192 with a source rate of 3.7 curies. The patient, a resident of a nearby nursing home, was returned there after the treatment and died there on Nov. 21, 1992. From materials found in the container of trash along with the radiation source, the NRC said it appeared that the source was disposed of with the nursing home's segregated biologically contaminated trash destined for incineration.

Preliminary information from NRC staff who investigated the incident is that the radioactive source apparently became detached from the cable that contained it. The cable is part of a mechanism used to insert the source into previously emplaced catheters in the patient's body. "The source apparently remained in the patient's body when she was returned to the nursing home," NRC said. An NRC Incident Investigation Team (IIT) was dispatched to the Indiana, Pa., area and was to visit other locations as deemed necessary.

The NRC said the licensee had agreed to (1) retain the source for NRC examination until further notice; (2) refrain from using the machinery for placing the source in patients until NRC gives written permission; (3) preserve the findings of the licensee's own investigation, determination of root causes of the incident, potential radiation exposures to the public and any involved workers, and preserve all evidence, equipment, and materials until receiving written permission to release them to others; and (4) notify every person found to have been involved in this incident of their possible exposure to radiation. The NRC staff worked with the treatment center, the nursing home, and the waste carrier to identify all persons who may have had contact with the patient over the several days after her treatment, before and after her death, as well as waste carrier employees who may have come in proximity to the source.

In mid-December 1992 the IIT leader issued a statement giving the team's findings not only concerning this incident but also a second one involving a similar machine.[14] The IIT found as a preliminary finding that in the patient's death in the Indiana case, shortly after the exposure, radiation appears to have been a potential contributing cause of her death.

The second machine failure, in Pittsburgh, appeared to have caused no injuries or consequences to the patient or the physician performing the treatment. In both instances the machines, duplicate models by the same manufacturer, used a thin metal cable inside a plastic tube (catheter) to insert a radioactive source in the tip of the cable into the patients' bodies. The machine is called a High Dose Rate afterloader brachytherapy machine. In both cases, the radioactive tip broke off the end of the cable. In the Indiana, Pa., case, the tip remained in the patient's body. In the Pittsburgh case, the tip had been withdrawn after treatment and was located in the catheter just outside the patient's body.

The IIT members and a contractor were sent to the facilities of Omnitron International, in Texas, the vendor of the brachytherapy device involved in the first event. The team also investigated a second failure of an Omnitron 2000 on Dec. 7, 1992, at the Greater Pittsburgh Cancer Center of Oncology Services Corporation. The president of Omnitron informed the IIT that the break on the second device was at the same location as on the first device (i.e., approximately at the interface between the source cavity of the wire and the cold portion of the wire).

In the second incident the patient was being treated with a single catheter in the lung. The break occurred as the source was being withdrawn from the lung into the remote afterloader. The source was out of the patient and near the connection of the catheter to the afterloader. The medical physicist operating the device received an alarm from the device indicating that the source had not fully retracted. He entered the treatment area with a Geiger counter; he saw the source in the catheter and immediately cut the catheter between the source and the patient and removed the patient from the room. He then went back and placed the source into a lead container next to the afterloader. Room radiation levels dropped to normal levels. Film badge readings for the Indiana Regional Cancer Center staff involved in the first event on November 16 ranged from 0.11 to 0.82 rem.

The ITT reported that

> Although the cause of the source break is still not known, the IIT has made the following preliminary findings for the failure of the Indiana Regional Cancer Center staff to detect the source remaining in the patient's body: (1) Wire breakage was not considered a credible accident by the Center staff. (2) Although a radiation monitor in the treatment room was alarming, it was disregarded. Some staffers considered it unreliable because it had alarmed in the past when no radiation was present. (3) The staff gave more credence to the Omnitron control computer which

they stated showed the source fully retracted and which they considered reliable. The only problem they believed they had was an inability to insert the source into the 5th and last catheter. Had the available Geiger counter been used to check the validity of the radiation monitor alarm in the treatment room the source would have been discovered and removed.

NRC PROPOSES REVISIONS TO REACTOR SITING REGULATIONS

The NRC proposed to amend its requirements governing the siting of nuclear power pants to decouple siting issues from those associated with reactor design and to take into account advancements in the earth sciences and earthquake engineering as they apply to the siting of nuclear power plants.[15]

As proposed, the revisions would

1. Eliminate the requirements to postulate accident source terms and for the use of dose calculations (these requirements would be retained for existing nuclear power plants and nonpower reactors).
2. Require a minimum exclusion area of 0.4 mile.
3. Establish population density criteria for use in assessing the suitability of future nuclear power plant sites. (As proposed, the population density at the time of initial site approval should not exceed 500 people per square mile averaged over any radial distance out to 30 miles. Forty years after initial site approval, the density should not exceed 1000 people per square mile out to a radial distance of 30 miles. If these population densities could be exceeded, consideration of alternative sites would be required, but they would not constitute upper limits of acceptability because severe accident risk considerations show that low risk can be achieved for sites having significantly higher population densities.)
4. Require that reviews of applications for early site approvals take into account important factors, such as population distribution, topography, and transportation routes, to determine whether there are any site characteristics that could pose a significant impediment to the development of an off-site emergency plan, such as limitations of access or egress in the immediate vicinity of the proposed site.
5. Update the seismic siting and engineering criteria for new nuclear power plants to benefit from the rapid advancement in the state of the art of earth sciences and the experience gained in the application of the procedures and methods used in the current regulation.

In 1976, the Public Interest Research Group filed a petition for rulemaking asking the Commission to establish minimum exclusion-area and low-population-zone distances and population density limits. The following year, Free Environment, Inc., and others filed a petition for rulemaking requesting, among other things, that the Commission require that the central Iowa nuclear project and other reactors be sited at least 40 miles from major population centers.

In response, the Commission, in 1978, directed its staff to develop a policy statement on nuclear power plant siting and a resulting report, "Report of the Siting Policy Task Force," was issued in 1979 and provided the staff's recommendations. In July 1980 the Commission issued an Advance Notice of Proposed Rulemaking regarding the staff's recommendations and seeking public comments on the matter. The proposed rulemaking was deferred the following year, however, to await development of the Safety Goal and improved research on accident source term.

FIRST-OF-A-KIND ENGINEERING PLANNED FOR ADVANCED LWRs

On Nov. 10, 1992, NRC staff met with the Nuclear Management Resources Council (NUMARC) to discuss the first-of-a-kind engineering (FOAKE) planned for standardized advanced light-water reactors (ALWRs). The purposes of the meeting were to provide an information exchange on the FOAKE approach and to discuss the potential issues arising under the FOAKE process that might need early consideration. The goals of the FOAKE program are to complete the engineering of certified designs of standard ALWR plants in sufficient detail to define the cost estimates and prepare for construction as well as to define the process to achieve commercial standardization.[16]

The two issues discussed at this meeting were (1) piping design improvements and (2) seismic equipment qualification. The NUMARC discussed the need for piping design improvements in ALWR plants because in its view the *ASME Boiler and Pressure Vessel Code* (ASME Code), Section III design requirements for piping are overly conservative and do not reflect current technical knowledge. The NRC staff noted that several changes to the NRC regulations and guidelines for piping design have already been proposed for implementation in the ALWR lead plant piping design. These changes include the elimination of the operating basis earthquake from design and an increase in the functional capability stress limits for piping. The NRC staff requested that NUMARC assess the reasonableness of the total package of piping design criteria proposed for the ALWR lead plant and its impact on piping system design before it proposes to reduce the safety margin further.

The NRC staff further pointed out that the piping methods used in the FOAKE must be consistent with the methods described in standard plant design certification. If changes to the ASME Code piping criteria are proposed for FOAKE, it might take a long time before those changes are adopted by the Code and endorsed by the NRC. The time factor could be a deterrent to the incorporation of the proposed changes into the design certification process for the ALWR evolutionary plants, according to NRC.

The NUMARC discussed its proposed approach for the seismic qualification of safety-related equipment. It intends to develop guidelines for equipment seismic qualification that would encompass all qualification methodologies (i.e., analyses, tests, and experience). The proposed approach for equipment seismic qualification would use traditional seismic qualification methods on equipment types with the highest seismic uncertainty, but it would allow the use of lessons learned from actual earthquake experience. With respect to an experience-based approach, NUMARC noted that there is a need for more definitive guidance on how this approach may be used on a case-by-case basis.

On the basis of the information currently submitted by the ALWR vendors for design certification, the NRC staff noted that the proposed approach might not be viable for FOAKE at this time. The staff stated that it would re-review the design certification applications to assess whether the description of the equipment seismic qualification methods would encompass such an approach. NUMARC also agreed to discuss this approach with the ALWR vendors to confirm whether they believe such an approach is encompassed by their submittal.

GAO REPORTS ON NUCLEAR REACTORS IN CUBA

In response to a request made by Congressman Robert Graham (D-FL), the GAO has issued a report providing information on the status of the construction of two Soviet-designed nuclear power reactors in Cuba as well as on allegations by former Cuban nuclear power officials that poor construction practices and other problems could affect the plant's operation.[17]

The report also discusses concerns of officials from the U.S. State Department, the NRC, and the DOE about the safety of the Cuban nuclear power reactors. It further presents information from the U.S. Geological Survey on the potential for earthquakes at the reactor site and from the National Oceanic and Atmospheric Administration on the probability that radioactive pollutants accidentally released into the atmosphere from the Cuban nuclear reactors could reach the United States.

Background

In 1976, the Soviet Union and Cuba concluded an agreement to construct two 440-MW nuclear power reactors near Cienfuegos on the south central coast of Cuba, about 180 miles south of Key West, Fla. The construction of these reactors, which began around 1983, had high priority in Cuba because of the country's heavy dependence on imported oil. Cuba is estimated to need an electrical generation capacity of 3000 MW by the end of the decade, GAO said. When completed, the first reactor unit would provide a significant percentage (estimated at over 15%, according to GAO) of Cuba's need for electricity.

Most of the reactor parts, except for civil construction materials, were supplied by the former Soviet Union under bilateral economic cooperation agreements. Cuba originally had planned to start up the first reactor by the end of 1993, but construction lags, technical complications, and problems with deliveries of equipment caused delays, according to the GAO report. Because of the breakup of the Soviet Union, Russian economic links with Cuba were disrupted when the newly formed Russian republic shifted to a market economy and began to place technical assistance to Cuba on a commercial basis. The GAO found that these changes contributed to the delays in the operational starting date for the reactors.

Design of Cuban Reactors

Cuba's nuclear power reactors are the newest design models (VVER-440 model) of the Soviet-designed 440-MW PWRs and are the first Soviet-designed reactors to be built in the Western Hemisphere and in a tropical environment. The Cuban model, called the VVER-440 V318, is the model that the Soviet Union planned to export to other countries. The most notable difference between the Cuban model and other Soviet-designed reactors, according to GAO, is that the Cuban reactors will have a full containment. The containment, a steel-lined concrete dome-like structure, serves as the ultimate barrier to a release of radioactive material in the event of a severe accident.

Study of Cuban Reactors

Because of Cuba's proximity to the United States, the NRC performed a limited study to examine the containment design and safety features of the Cuban nuclear power reactors. The study was completed in 1989 and discusses similarities and differences in safety characteristics between the Cuban reactors and comparable U.S. reactors.

The study noted that, although the design of the Cuban reactors has many features in common with that of the U.S. PWR, several differences could lead to significantly different reactions in the event of a serious accident. For example, the Cuban reactor, like the U.S. PWR, uses water to cool the reactor core, but the Cuban reactor uses a different system for handling the steam pressure that would be generated by a severe accident. In the Cuban reactor design, the steam is condensed to water in a bubbler–condenser system so that pressure is reduced in the containment structure, the NRC report noted. The NRC determined that if, in a worst-case scenario, the steam bypassed the bubbler–condenser system and reached the upper portion of the containment in pressures greater than the upper portion's designed pressure retention capability of 7 pounds per square inch, the containment could be breached, and a radioactive release could occur. In contrast, U.S. PWRs are designed to accommodate pressures of about 50 pounds per square inch throughout the entire containment structure. The NRC study indicated that the Cuban reactor and comparable U.S. PWRs are designed to accommodate similar types of accidents but concluded that it was difficult to compare the risk posed by the two types of reactors because the information required for such an assessment was not available.

Status of Construction

On Sept. 5, 1992, Cuban Chairman Fidel Castro announced that the construction of both of Cuba's reactors was being suspended because Cuba could not meet the financial terms set by the post-Soviet Russian government to complete the reactors, the GAO report states. Estimates of the amount of the civil construction completed for the first nuclear power reactor range from 90 to 97%, but only about 37% of the reactor equipment had been installed. About 20 to 30% of the civil construction is estimated to be completed for the second reactor, but no information was available about the status of the equipment for the second reactor.

The GAO reported that concrete has been poured on the upper portion of the containment dome for the first unit but that the reactor's I&C system had not been purchased because Cuba did not have the hard currency to pay for it. The reactor fuel has not been delivered, and some key or primary system components (one reactor vessel, six steam generators, five primary coolant pumps, twelve isolation valves, one pressurizer, a catch tank, and four accumulators) have been in outdoor storage on site since December 1990.

According to information provided to GAO by an official of the Embassy of the Russian Federation in Washington, DC, the first nuclear reactor was tentatively scheduled to be operational in late 1995 or early 1996. Because Cubans constructing the reactor lack experience, all critical work was being done by Russians or under the control of the Russians, GAO reported. As of Apr. 1, 1992, the cost of the plant's construction totaled 1.6 billion rubles, or about $960 million.

Safety Concerns of Former Officials

The GAO interviewed five former Cuban nuclear power officials who were identified as having concerns about the Cuban reactors. These officials included nuclear and electrical engineers and a technician who had worked at the reactor site and emigrated from Cuba. The GAO said the officials believe that problems exist that could affect the safe operation of the reactors, such as the lack of a system to check reactor components, defective welds in the civil construction, and questionable training of future operators.

According to the former Cuban nuclear power officials, the nuclear facility does not have a good system to check reactor components. A former Cuban technician, who was responsible for checking welds in the civil construction, told GAO that he and a Soviet technician had examined X-rays from about 5000 weld sites that had passed inspection. They found that between 10 and 15% of these welds were defective. Although he did not know exactly where the pipes with the defective welds were located, it was thought that they were a part of the auxiliary plumbing system. According to this former technician, a group of Soviet officials also reviewed the X-rays and confirmed that the welds were defective. Another former official said that even though defective welds were found in the containment dome, concrete was still poured.

In June 1991 this former Cuban official testified on problems in the reactor's civil construction before the Subcommittee on Western Hemisphere Affairs of the House Committee on Foreign Affairs. State Department, DOE, and NRC officials debriefed this individual and concluded that the Cuban reactors appeared to have quality control problems but that the welding problems probably would not lead to a major accident, GAO reported. Two of the former Cuban officials who were still working at the nuclear power plant at the time of the hearings told GAO the Cubans had paid increased attention to safety concerns after this individual testified.

According to another former Cuban official, individuals being trained to be Cuban reactor operators received 5 months of instruction from Russians on a VVER-440 MW model V230 reactor simulator at the Novovoronezh nuclear power plant in Russia. He said, however, that the value of this training is questionable because this simulator does not resemble the reactor under construction in Cuba.

The Acting Principal Officer of the Cuban Interests Section of the State Department told the GAO that he was aware of the allegations made by the Cuban emigres. He said, however, that Cuba was interested in building the nuclear reactor in accordance with recognized safety standards to avoid the effects that a "Chernobyl-type" accident could have on Cuban people and surrounding countries. He added that he did not know whether the plant would ever be finished because so much money was needed to buy equipment for the reactors.

The GAO submitted a list of written questions to this official about the status of the reactors' construction, design, and operational safety features and nuclear fuel. He told GAO that he would submit the questions to the appropriate nuclear power officials in his government and try to arrange for GAO staff to meet with Cuban nuclear power officials and visit the nuclear plant site. As of September 1, GAO had not received a response to the questions.

United States Prefers Reactors not be Completed

Currently, the United States maintains a comprehensive embargo on any U.S. transactions with Cuba and discourages other countries from providing assistance, except for safety purposes, to Cuba's nuclear program. The United States would prefer that the construction of the reactors not be completed and insists that Cuba sign either the Non-Proliferation Treaty or the Treaty of Tlatelolco—both of which bind signatories to blanket nonproliferation commitments for their entire nuclear program—before the United States could consider reversing its policy of discouraging other countries from assisting Cuba with the construction of the reactors, GAO said.

According to the State Department, U.S. nuclear officials believed, on the basis of information available about the design of the power plant, that the possibility of an off-site radiation leak was considerably lower for the Cuban reactors than for "Chernobyl-type" reactors because of design differences. However, U.S. officials are concerned that Cuba is not equipped to deal with an accident, according to GAO.

The NRC is also concerned. According to NRC's director of international programs, however, before NRC could form an opinion on Cuba's nuclear reactors, a team of NRC inspectors and/or U.S. nuclear industry officials would have to conduct an extensive investigation of the plant and be given access to information about construction procedures, techniques, and test results. Such a team would also need to inspect construction and equipment installation visually as they occur. The director expressed concern about the design of the plant's containment system, specifically the design of the pressure suppression system.

Assessments of Risk From Earthquakes and Radioactive Pollutants

United States Geological Survey officials could not determine the potential for earthquakes at the reactor site, in part because available information was limited. At the request of GAO, however, the Office of International Geology analyzed, by season, the probability of impact, the average arrival time, and the relative concentrations of radioactive pollutants that would be released into the atmosphere by an accidental release of radioactivity from the Cuban reactors. On the basis of climatological data for the summer of 1991 and the winter of 1991–1992, the analysis showed that the summer east-to-west trade winds could carry radioactive pollutants over all of Florida and portions of the Gulf states as far west as Texas in about 4 days. In winter, when trade winds are weaker and less persistent, radioactive pollutants would encounter strong westerly winds that could move the pollutants toward the east, possibly as far north as Virginia and Washington, DC, in about 4 days.

REFERENCES

1. Letter to I. Selin, Chairman, U.S. Nuclear Regulatory Commission, from D. A. Ward, Chairman, Advisory Committee on Reactor Safeguards, *Digital Instrumentation and Control System Reliability*, Sept. 16, 1992.
2. Letter to J. Taylor, Executive Director of Operations, U.S. Nuclear Regulatory Commission, from D. A. Ward, Chairman, Advisory Committee on Reactor Safeguards, *Draft Commission Paper, "Design Certification and Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light Water Reactor Designs,"* Sept. 16, 1992.
3. Letter to J. Taylor, Executive Director of Operations, U.S. Nuclear Regulatory Commission, from D. A. Ward, Chairman, Advisory Committee on Reactor Safeguards, *General Electric Nuclear Energy Power Uprate Program/Fermi, Unit 2 Power Increase Request*, Sept. 17, 1992.
4. Letter to J. Taylor, Executive Director of Operations, U.S. Nuclear Regulatory Commission, from D. A. Ward, Chairman,

Advisory Committee on Reactor Safeguards, *Proposed Guidance for Implementation of the Maintenance Rule, 10 CFR 50.65,* cited in NRC Press Release 92-162 (Nov. 4, 1992).

5. Letter to J. Taylor, Executive Director of Operations, U.S. Nuclear Regulatory Commission, from D. A. Ward, Chairman, Advisory Committee on Reactor Safeguards, *Proposed Branch Technical Position on Environmental Qualification of Electrical Equipment for License Renewal,* cited in NRC Press Release 92-162 (Nov. 4, 1992).
6. Letter to J. Taylor, Executive Director of Operations, U.S. Nuclear Regulatory Commission, from D. A. Ward, Chairman, Advisory Committee on Reactor Safeguards, *Proposed Final Amendments to 10 CFR Part 55 on Renewal of Licenses and Requalification,* cited in NRC Press Release 92-162 (Nov. 4, 1992).
7. Letter to I. Selin, Chairman, U.S. Nuclear Regulatory Commission, from P. Shewmon, Chairman, Advisory Committee on Reactor Safeguards, *Environmental Qualification for Digital Instrumentation and Control Systems,* cited in NRC Press Release 92-168 (Nov. 24, 1992).
8. Letter to I. Selin, Chairman, U.S. Nuclear Regulatory Commission, from P. Shewmon, Chairman, Advisory Committee on Reactor Safeguards, *Risk-Based Regulation,* cited in NRC Press Release 92-168 (Nov. 24, 1992).
9. *At. Energy Clearing House,* 38(40): 5 (Oct. 2, 1992).
10. *At. Energy Clearing House,* 38(41): 3 (Oct. 9, 1992).
11. *At. Energy Clearing House,* 38(45): 11 (Nov. 6, 1992).
12. *At. Energy Clearing House,* 38(48): 13 (Nov. 25, 1992).
13. *At. Energy Clearing House,* 38(49): 3 (Dec. 4, 1992).
14. *At. Energy Clearing House,* 38(51): 1 (Dec. 18, 1992).
15. *At. Energy Clearing House,* 38(45): 3 (Nov. 6, 1992).
16. *At. Energy Clearing House,* 38(49): 6 (Dec. 4, 1992).
17. *At. Energy Clearing House,* 38(50): 3 (Dec. 11, 1992).

Reports, Standards, and Safety Guides

By D. S. Queener[a]

This article contains four lists of various documents relevant to nuclear safety as compiled by the editor. These lists are: (1) reactor operations-related reports of U.S. origin, (2) other books and reports, (3) regulatory guides, and (4) nuclear standards. Each list contains the documents in its category which were published (or became available) during the three-month period (October, November, and December 1992) covered by this issue of *Nuclear Safety*. The availability and cost of the documents are noted in most instances.

OPERATIONS REPORTS

This category is listed separately because of the increasing interest in the safety implications of information obtainable from both normal and off-normal operating experience with licensed power reactors. The reports fall into several categories shown, with information about the availability of the reports given where possible. The NRC reports are available from the Nuclear Regulatory Commission (NRC) Public Document Room, 2120 L Street, NW, Washington, DC 20555.

NRC Office of Nuclear Reactor Regulation

The NRC Office of Nuclear Reactor Regulation (NRR) issues reports regarding operating experience at licensed reactors. These reports, previously published by the NRC Office of Inspection and Enforcement (IE), fall into two categories of urgency: (1) NRC Bulletins, which require remedial actions and/or responses from affected licensees, and (2) NRC Information Notices, which are for general information and do not require any response from the licensee. The NRR also periodically issues Generic Letters (GL) to licensees, usually for information purposes only.

NRC Information Notices

NRC IN 91-64, Supplement 1 *Site Area Emergency Resulting from a Loss of Non-Class 1E Uninterruptible Power Supplies,* October 7, 1992, 3 pages plus two pages of attachments.

NRC IN 92-72 *Employee Training and Shipper Registration Requirements for Transporting Radioactive Materials,* October 28, 1992, 4 pages plus one-page attachment.

NRC IN 92-73 *Removal of a Fuel Element from a Research Reactor Core While Critical,* November 4, 1992, 3 pages plus one-page attachment.

NRC IN 92-74 *Power Oscillations at Washington Nuclear Power Unit 2,* November 10, 1992, 5 pages plus 2 pages of attachments.

NRC IN 92-75 *Unplanned Intakes of Airborne Radioactive Material by Individuals at Nuclear Power Plants,* November 12, 1992, 4 pages plus one-page attachment.

NRC IN 92-76 *Issuance of Supplement 1 to NUREG-1358, "Lessons Learned from the Special Inspection Program for Emergency Operating Procedures (Conducted October*

[a]Oak Ridge National Laboratory.

1988–September 1992)," November 13, 1992, 3 pages plus one-page attachment.

NRC IN 92-77 *Questionable Selection and Review to Determine Suitability of Electropneumatic Relays for Certain Applications,* November 17, 1992, 3 pages plus one-page attachment.

NRC IN 92-78 *Piston to Cylinder Liner Tin Smearing on Cooper–Bessemer KSV Diesel Engines,* November 30, 1992, 3 pages plus one-page attachment.

NRC IN 92-79 *Non-Power Reactor Emergency Event Response,* December 1, 1992, 2 pages plus one-page attachment.

NRC IN 92-80 *Operation with Steam Generator Tubes Seriously Degraded,* December 7, 1992, 4 pages plus one-page attachment.

NRC IN 92-81 *Potential Deficiency of Electrical Cables,* December 11, 1992.

NRC IN 92-82 *Results of Thermo-Lag 330-1 Combustibility Testing,* December 15, 1992.

NRC IN 92-83 *Thrust Limits for Limitorque Actuators and Potential Overstressing of Motor-Operated Valves,* December 17, 1992, 3 pages plus 2 pages of attachments.

NRC IN 92-84 *Release of Patients Treated with Temporary Implants,* December 17, 1992.

NRC IN 92-85 *Potential Failures of Emergency Core Cooling Systems Caused by Foreign Material Blockage,* December 23, 1992.

Other Operations Reports

These are other reports issued by various organizations in the United States dealing with power-reactor operations activities. Most of the NRC publications (NUREG series documents) can be ordered from the Superintendent of Documents, U.S. Government Printing Office (GPO), P.O. Box 37082, Washington, DC 20013. NRC draft copies of reports are available free of charge by writing the NRC Office of Administration (ADM), Distribution and Mail Services Section, Washington, DC 20555. A number of these reports can also be obtained from the NRC Public Document Room (PDR). Specify the report number when ordering. Telephone orders can be made by contacting the PDR at (202) 634-3273.

Many other reports prepared by U.S. Government laboratories and contractor organizations are available from the National Technical Information Service (NTIS), Springfield, VA 22161, and/or DOE Office of Scientific and Technical Information (OSTI), P.O. Box 62, Oak Ridge, TN 37831. Reports available through one or more of these organizations are designated with the appropriate information (i.e., GPO, PDR, NTIS, and OSTI) in parentheses at the end of the listing, followed by the price, when available.

NUREG-1358, Supplement 1 *Lessons Learned from the Special Inspection Program for Emergency Operating Procedures, Conducted October 1988–September 1991,* October 1992, 29 pages (GPO).

NRC Office for Analysis and Evaluation of Operational Data

The NRC Office for Analysis and Evaluation of Operational Data (AEOD) is responsible for the review and assessment of commercial nuclear power plant operating experience. The AEOD publishes a number of reports, including case studies, special studies, engineering evaluations, and technical reviews. Individual copies of these reports can be obtained from the NRC Public Document Room (PDR).

AEOD/C92-01 *Case Study on Human Performance in Operating Events,* John V. Kauffman et al., December 1992, 50 pages.

AEOD/S92-07 *Special Study on Pressure Locking and Thermal Binding of Gate Valves,* C. Hsu, December 1992, 40 pages.

AEOD/T92-08 *Emergency Diesel Generator Start Frequency,* October 15, 1992, 5 pages.

AEOD/T92-09 *Review of Manual Valve Failures,* November 25, 1992, 19 pages.

AEOD/T92-10 *Prospective Trend of Low Reliability Emergency Diesel Generators,* December 1992, 7 pages.

DOE- and NRC-Related Items

NUREG/CR-4012 *Replacement Energy Costs for Nuclear Electricity-Generating Units in the United States: 1992–1996,* J. C. VanKuiken et al., Argonne National Lab., Ill., October 1992, 227 pages (GPO).

NUREG/CR-4832, Vol. 4 *Analysis of the LaSalle Unit 2 Nuclear Power Plants: Risk Methods Integration and Evaluation Program (RMIEP), Initiating Events and Accident Sequence Delineation,* A. C. Payne, Jr., et al., Sandia National Labs., N.M., October 1992, 105 pages (GPO).

NUREG/CR-5545, Rev. 1 *VICTORIA: A Mechanistic Model of Radionuclide Behavior in the Reactor Coolant System Under Severe Accident Conditions,* T. J. Heames et al., December 1992, 194 pages (GPO).

NUREG/CR-5826 *Auxiliary Feedwater System Risk-Based Inspection Guide for the Maine Yankee Nuclear Power Plant,* B. F. Gore et al., Pacific Northwest Lab., Wash., October 1992, 28 pages (GPO).

NUREG/CR-5932 *Risk-Based Inspection Guide for the Susquehanna Station HPCI System,* R. Travis et al., Brookhaven National Lab., N.Y., November 1992, 44 pages (GPO).

REGULATORY GUIDES

To expedite the role and function of the NRC, its Office of Nuclear Regulatory Research prepares and maintains a file of Regulatory Guides that define much of the basis for the licensing of nuclear facilities. These Regulatory Guides are divided into 10 divisions as shown in Table 1.

Table 1 Regulatory Guides

Division 1	Power Reactor Guides
Division 2	Research and Test Reactor Guides
Division 3	Fuels and Materials Facilities Guides
Division 4	Environmental and Siting Guides
Division 5	Materials and Plant Protection Guides
Division 6	Product Guides
Division 7	Transportation Guides
Division 8	Occupational Health Guides
Division 9	Antitrust and Financial Review Guides
Division 10	General Guides

Single copies of the draft guides may be obtained from NRC Distribution Section, Division of Information Support Services, Washington, DC 20555. Draft guides are issued free (for comment) and licensees receive both draft and final copies free; others can purchase single copies of active guides by contacting the U.S. Government Printing Office (GPO), Superintendent of Documents, P.O. Box 37082, Washington, DC 20013. Costs vary according to length of the guide. Of course, draft and active copies will be available from the NRC Public Document Room, 1717 H Street, NW, Washington, DC, for inspection and copying for a fee.

Revisions in these rates will be announced as appropriate. Subscription requests should be sent to the National Technical Information Service, Subscription Department, Springfield, VA 22161. Any questions or comments about the sale of regulatory guides should be directed to Chief, Document Management Branch, Division of Technical Information and Document Control, Nuclear Regulatory Commission, Washington, DC 20555.

Actions pertaining to specific guides (such as issuance of new guides, issuance for comment, or withdrawal), which occurred during the October, November, and December 1992 reporting period, are listed below.

Division 1 Power Reactor Guides

1.012 (Draft revision 2) *Nuclear Power Plant Instrumentation for Earthquakes,* November 1992.

1.028 (Proposed revision 4) *QA Program Requirements,* November 1992.

Division 4 Environmental and Siting Guides

4.007 (Draft revision 2) *General Site Suitability Criteria for Nuclear Power Stations,* November 1992.

Proposed Rule Changes as of Dec. 31, 1992[a,b]

(Changes Since the Previous Issue of *Nuclear Safety* Are Indicated by Shaded Areas)

Number of part to be changed	Date published for comment	Date comment period expired	Date published; date effective	Topic or proposed effect	Current action and/or comment, *Federal Register* volumes and page numbers
10 CFR 1	2-4-92	5-4-92		Elimination of requirements marginal to safety	Published for comment in 57:23 (4166)
10 CFR 1	2-24-92	3-6-92		Special review of NRC regulations	Published for comment in 57:36 (6299)
10 CFR 1	6-19-92	8-18-92; 9-30-92		Review of reactor licensee reporting requirements	Published for comment in 57:119 (27394); comment period extended in 57:153 (34886)
10 CFR 2	10-24-90	12-10-90		Options and procedures for direct Commission review of licensing board decisions	Published for comment in 55:206 (42947)
10 CFR 2	12-23-92	3-8-93		Availability of official records	Published for comment in 57:247 (61013)
10 CFR 11 10 CFR 19 10 CFR 20 10 CFR 21 10 CFR 25 10 CFR 26 10 CFR 30 10 CFR 31 10 CFR 32 10 CFR 33 10 CFR 34 10 CFR 35 10 CFR 39 10 CFR 40 10 CFR 50 10 CFR 52 10 CFR 53 10 CFR 54 10 CFR 55 10 CFR 60 10 CFR 61 10 CFR 70 10 CFR 71 10 CFR 72 10 CFR 73 10 CFR 74 10 CFR 75 10 CFR 95 10 CFR 110 10 CFR 140 10 CFR 150	1-3-92	3-18-92	11-24-92; 12-24-92	Clarification of statutory authority for purposes of criminal enforcement	Published for comment in 57:2 (222); final rule in 57:227 (55062)

Proposed Rule Changes as of Dec. 31, 1992 (Continued)

Number of part to be changed	Date published for comment	Date comment period expired	Date published; date effective	Topic or proposed effect	Current action and/or comment, *Federal Register* volumes and page numbers
10 CFR 11 10 CFR 25			9-10-92; 10-13-92	Access authorization fee schedule for licensee personnel	Final rule in 57:176 (41375); correction in 57:229 (56406)
10 CFR 19 10 CFR 20 10 CFR 21 10 CFR 30 10 CFR 36 10 CFR 40 10 CFR 51 10 CFR 70 10 CFR 170	12-4-90	3-4-91		Licenses and radiation safety requirements for large irradiators	Published for comment in 55:233 (50008)
10 CFR 19			12-29-92; 3-1-93	Exclusion of attorneys from interviews under subpoena	Final rule in 57:250 (61780)
10 CFR 20 10 CFR 61	4-21-92	7-20-92		Low-level waste shipment manifest information and reporting	Published for comment in 57:77 (14500)
10 CFR 20			12-7-92; 1-6-93	Disposal of waste oil by incineration	Final rule in 57:235 (57649)
19 CFR 20			12-8-92; 12-8-92	Revised standards for protection against radiation; minor amendments	Final rule in 57:236 (57877)
10 CFR 26 10 CFR 70 10 CFR 73	4-30-92	7-29-92		Fitness-for-duty requirements for licensees who possess, use, or transport Category I material	Published for comment in 57:84 (18415); correction in 57:101 (22021)
10 CFR 26			11-25-92; 12-28-92	Fitness-for-Duty programs: NRC partial withdrawal of NRC information collection requirements	Final rule in 57:228 (55443)
10 CFR 30 10 CFR 40 10 CFR 70 10 CFR 72	10-7-91	12-23-91		Decommissioning recordkeeping and license termination: documentation	Published for comment in 56:194 (50524)
10 CFR 30 10 CFR 40 10 CFR 70	2-20-92	4-30-92		Proposed method for regulating major materials licenses; availability of NUREG report	Published for comment in 57:34 (6077)
10 CFR 30 10 CFR 35	6-11-92	7-13-92	10-2-92; 10-2-92	Departures from manufacturer's instructions; elimination of recordkeeping requirements	Published for comment in 57:113 (24763); final rule in 57:192 (45566)
10 CFR 31 10 CFR 32	12-27-91	3-12-92		Requirements for the possession of industrial devices containing byproduct material	Published for comment in 56:248 (67011)

(Table continued on the next page)

Proposed Rule Changes as of Dec. 31, 1992 (Continued)

Number of part to be changed	Date published for comment	Date comment period expired	Date published; date effective	Topic or proposed effect	Current action and/or comment, *Federal Register* volumes and page numbers
10 CFR 31 10 CFR 32	11-27-92	3-29-93		Requirements concerning the accessible air gap for generally licensed devices	Published for comment in 57:229 (56287)
10 CFR 34 10 CFR 35 10 CFR 50 10 CFR 73 10 CFR 110			12-29-92; 12-29-92	Material approved for incorporation by reference; maintenance and availability	Final rule in 57:250 (61785)
10 CFR 40	10-28-92	1-26-93		Licensing of source material	Published for comment in 57:209 (48749)
10 CFR 48 (Chap. 20)			12-23-92; 1-22-93	Acquisition regulation (NRCAR)	Final rule in 57:247 [Part II] (61152)
10 CFR 50 10 CFR 52	1-7-92	3-9-92		Training and qualification of nuclear power plant personnel	Published for comment in 57:4 (537)
10 CFR 50	4-21-92	7-6-92		Loss of all alternating current power	Published for comment in 57:77 (14514)
10 CFR 50	4-24-92	7-8-92	10-21-92; 11-20-92	Receipt of byproduct and special nuclear material	Published for comment in 57:80 (15034); final rule in 57:204 (47978)
10 CFR 50	6-26-92	7-27-92		Minor modifications to nuclear power reactor event reporting requirements	Published for comment in 57:124 (28642); final rule in 57:176 (41376)
10 CFR 50	9-28-92	12-28-92		Acceptability of plant performance for severe accidents; scope of consideration in safety regulations	Published for comment in 57:188 (44513)
10 CFR 50 10 CFR 52 10 CFR 100	10-20-92	2-17-93		Reactor site criteria, including seismic and earthquake engineering criteria for nuclear power plants and proposed denial of petition for rulemaking from Free Environment, Inc., et al.	Published for comment in 57:203 (47802)
10 CFR 51	9-17-91	12-16-91; 3-16-92		Environmental review for renewal of operating licenses	Published for comment in 56:180 (47016); comment period extended in 56:228 (59898)
10 CFR 51	7-23-90	10-22-90		License renewal for nuclear power plants; scope of environmental effects	Advanced notice of proposed rulemaking published for comment in 55:141 (29964)
10 CFR 52		2-22-93	12-23-92; 1-22-93	Combined construction permits and operating licenses; conforming amendments	Final rule in 57:247 (60975)

Proposed Rule Changes as of Dec. 31, 1992 (Continued)

Number of part to be changed	Date published for comment	Date comment period expired	Date published; date effective	Topic or proposed effect	Current action and/or comment, *Federal Register* volumes and page numbers
10 CFR 55	4-17-90	7-2-90		Operator's licenses	Published for comment in 55:74 (14288)
10 CFR 61	3-6-92	4-6-92		Licensing requirements for land disposal of radioactive wastes	Published for comment in 57:45 (8093)
10 CFR 72	6-26-92	9-9-92		List of approved spent fuel storage casks: additions	Published for comment in 57:124 (28645)
10 CFR 73	12-13-91	3-13-92		Physical fitness programs and day firing qualifications for security personnel at Category 1 license fuel cycle facilities	Published for comment in 56:240 (65024)
10 CFR 73	5-29-92	8-12-92		Clarification of physical protection requirements at fixed sites	Published for comment in 57:104 (22670)
10 CFR 110	2-7-90	3-9-90		Import and export of radioactive wastes	Advance notice of proposed rulemaking for comment in 55:26 (4181); corrections in 55:57 (10786);
	4-28-92	7-13-92			published for comment in 57:82(17859)
48 CFR 20	10-2-89	12-1-89		Acquisition regulation (NRCAR)	Published for comment in 54:189 (40420)

[a]NRC petitions for rule making are not included here, but quarterly listings of such petitions can be obtained by writing to Division of Rules and Records, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Quarterly listings of the status of proposed rules are also available from the same address.

[b]Proposed rules for which the comment period expired more than 2 years prior to the start of the period currently covered without any subsequent action are dropped from this table. Effective rules are removed from this listing in the issue after their effective date is announced.

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Aluminum–Uranium Fuel-Melt Behavior During Severe Nuclear Reactor Accidents

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R&D Activities on Safety Aspects of Future PWR Plants Performed at KfK

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Upgrading of the K-Reactor Supplementary Safety System at the Savannah River Site

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VPBER-600 Conceptual Features and Safety Analysis Results

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Radiological Consequence Analyses Under Severe Accident Conditions for the Advanced Neutron Source Reactor at the Oak Ridge National Laboratory

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Summary of Fuel Performance Annual Report for 1990

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Reviewers of *Nuclear Safety,* Volume 34

The technical quality of a journal depends not only on the competence and efforts of its authors and editorial staff but also, to a major extent, on the dedication of its corps of peer reviewers. We wish to acknowledge gratefully the many technical experts whose voluntary and unrewarded reviews of proposed *Nuclear Safety* articles have been indispensable in the selection of articles and in the revision of articles to prepare them for publication.

We list below all the names of those who reviewed articles for publication in Vol. 34, whether the articles were used or not. Since it is our policy not to reveal the reviewers' identities to the authors, all reviewers are listed in alphabetical order together with their affiliations.

This list does not include, though we are most grateful to them also, the names of the DOE and NRC staff members who review all *Nuclear Safety* articles to ensure that the policies and positions of their agencies are not misstated or distorted.

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THIRD INTERNATIONAL CONFERENCE ON CONTAINMENT DESIGN AND OPERATION

Toronto, Canada, October 19–21, 1994

This conference is to be hosted by the Canadian Nuclear Society with co-sponsorship by various international organizations.

There are a number of specific international meetings on various aspects of containment. They do not cover all major important developments in this field in a single forum. The scope of this third international containment conference will provide such a forum and will cover all major aspects from design and analysis through commissioning and operation to aging. The conference will provide a colloquium to initiate and enhance dialogue between researchers, reactor operators, analysts, and licensing experts from various organizations and nations. A major opportunity for experts from east and west to contemplate solutions to containment issues is anticipated.

Topics to be addressed include performance and regulatory requirements for containment; separate effect verification and global validation of containment thermal–hydraulic and radionuclide behavior codes; operation, maintenance, leakage, and aging of containment systems; thermal–hydraulic behavior of containment systems; containment design for severe accidents and severe structural loading; radionuclide behavior in containment—experiment and analysis; containment passive safety systems design and operation; aerosol behavior in containment; containment reliability, integrity, and risk assessment; and hydrogen mixing, transport, burn, and detonation in containment.

300- to 500-word abstracts should be submitted in triplicate by November 15, 1993, to the following address, from which additional information may also be obtained; Containment Design and Operation Conference, The Canadian Nuclear Society, 144 Front Street West, Suite 725, Toronto, Ontario, Canada, M5G 2L7. Telephone: (416) 977-7620. FAX: (416) 979-8356.

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