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Proceedings of the U.S. Nuclear Regulatory Commission

Twenty-Fifth Water Reactor Safety Information Meeting

Volume 2
- Human Reliability Analysis and Human Performance Evaluation
- Technical Issues Related to Rulemakings
- Risk-Informed, Performance-Based Initiatives
- High Burn-up Fuel Research

Held at
Bethesda Marriott Hotel
Bethesda, Maryland
October 20-22, 1997

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

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ABSTRACT

This three-volume report contains papers presented at the Twenty-Fifth Water Reactor Safety Information Meeting held at the Bethesda Marriott Hotel, Bethesda, Maryland, October 20-22, 1997. The papers are printed in the order of their presentation in each session and describe progress and results of programs in nuclear safety research conducted in this country and abroad. Foreign participation in the meeting included papers presented by researchers from France, Japan, Norway, Russia, Spain and Switzerland. The titles of the papers and the names of the authors have been updated and may differ from those that appeared in the final program of the meeting.

PROCEEDINGS OF THE
25TH WATER REACTOR SAFETY INFORMATION MEETING

OCTOBER 20-22, 1997

Published in Three Volumes

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- Structural Performance

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An Experimental Investigation of the Effects of Alarm Processing and Display on Operator Performance

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Abstract: This paper describes a research program sponsored by the U.S. Nuclear Regulatory Commission to address the human factors engineering (HFE) aspects of nuclear power plant alarm systems. The overall objective of the program is to develop HFE review guidance for advanced alarm systems. As part of this program, guidance has been developed based on a broad base of technical and research literature. In the course of guidance development, aspects of alarm system design for which the technical basis was insufficient to support complete guidance development were identified. The primary purpose of the research reported in this paper was to evaluate the effects of three of these alarm system design characteristics on operator performance in order to contribute to the understanding of potential safety issues and to provide data to support the development of design review guidance in these areas. Three alarm system design characteristics studied were (1) alarm processing (degree of alarm reduction), (2) alarm availability (dynamic prioritization and suppression), and (3) alarm display (a dedicated tile format, a mixed tile and message list format, and a format in which alarm information is integrated into the process displays). A secondary purpose was to provide confirmatory evidence of selected alarm system guidance developed in an earlier phase of the project. The alarm characteristics were combined into eight separate experimental conditions. Six, two-person crews of professional nuclear power plant operators participated in the study. Following training, each crew completed 16 test trials which consisted of two trials in each of the eight experimental conditions (one with a low-complexity scenario and one with a high-complexity scenario). Measures of process performance, operator task performance, situation awareness, and workload were obtained. In addition, operator opinions and evaluations of the alarm processing and display conditions were collected. No deficient performance was observed in any of the experimental conditions, providing confirmatory support for many design review guidelines. The operators identified numerous strengths and weaknesses associated with individual alarm design characteristics.

I. INTRODUCTION

The need to improve the human factors engineering (HFE) of alarm systems has led to the development of advanced systems in which alarm data are processed beyond the traditional "one sensor - one alarm" framework. While this technology promises to provide a means of correcting many known alarm system deficiencies, there is also the potential to negatively impact operator performance [1]. A research program, sponsored by the U.S. Nuclear Regulatory Commission (NRC), is underway to address the HFE aspects of nuclear power plant alarm systems. The objective of the study is develop HFE review guidance for advanced, computer-based alarm systems. As part of the development effort, aspects of alarm design for which the technical basis was insufficient to support guidance development were identified and research to address the most significant issues was initiated.

This paper will present the current status of the program. Section II will provide an overview of our approach to guidance development and discuss the role of simulation in the methodology. In Section III, the current experimental research will be described to illustrate how the alarm system design features are being studied. The conclusions are presented on Section IV.

II. DEVELOPMENT OF ALARM SYSTEM REVIEW GUIDANCE

A general methodology was established to develop HFE guidance to support the NRC's review of NPP HSIs [2]. The methodology has been applied to several areas of new HSI technology and the guidance has been integrated into NUREG-0700, Revision 1 [3]. Guidance development proceeds as shown in Fig. 1. The methodology seeks to establish valid guidelines in a cost-effective manner. Validity is defined along two dimensions. "Internal" validity is the degree to which the individual guidelines are based upon an auditable research trail. "External" validity is the degree to which the guidelines are subjected to independent peer review. The peer review process is considered a good method of screening guidelines for conformance to accepted human engineering practices. Validity can be inherited from the source materials that are used to develop the guidelines. Thus, for example, for a specific topic there are sometimes existing documents, such as industry guidance documents and standards, that have an auditable research trail and have been the subject of extensive peer review. We refer to these as primary source documents. Where source materials lack validity, it must be establish for the new guidance as part of the guidance development process itself.

Since they already possess internal and external validity, primary source documents are sought first in our approach to guidance development. Even when such a documents are available, their guidance must still be adapted to an NPP HSI application. When primary source documents alone do not provide a sufficient basis on which to develop guidelines, additional sources of information are necessary. Secondary source documents are those with either internal or external validity (not both). Many industry guidance documents fit into this category. They are good from the standpoint that their information is already expressed in guideline format. However, they either provide a good trail to their technical basis *or* have been peer reviewed, so the missing aspect of validity needs to be established as part of the design review guidance development. Tertiary documents, such as HFE handbooks, generally do not provide information in guidance form and they do not possess either form of validity. Thus considerable effort may be involved in guideline preparation and validation using these sources.

The three final sources of information for guidance development (see Fig. 1) require the most effort. Basic literature and industry experience are used where guidelines cannot be obtained from the other sources. Results are evaluated from basic literature including articles from refereed technical journals, reports from research organizations, and papers from technical conferences. Industry experience can be obtained from surveys and interviews. Industry experience is a valuable information source of information for identifying performance issues associated with actual systems and tested design solutions to problems that have been resolved.

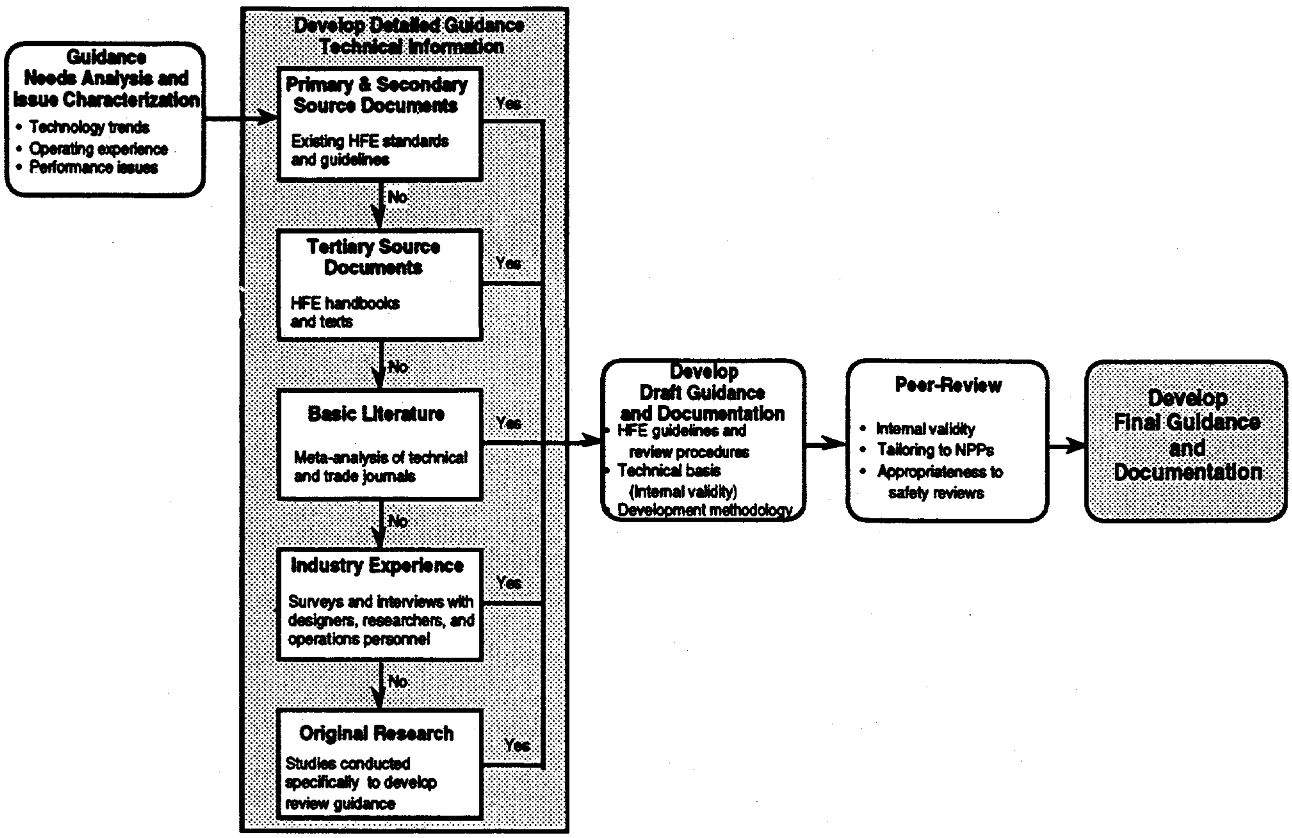

Fig. 1. Guidance development methodology.

Original research is the last category and refers to the systematic manipulation of the HSI design features of interest in order to determine their effects on performance under controlled conditions. The research should generally be performed in a dynamic, real-time context; e.g., a full-scope simulator or high-fidelity engineering simulator. This type of research plays two important roles in guidance development: technical basis development and guidance confirmation. First, when the technical basis does not exist in the other source materials the results of experiments can be used to fill the knowledge gap, i.e., to provide the information upon which design review guidance can be developed. For example, such studies can identify what aspects of system design that are significant to human performance.

The second important role of experimental research is guidance confirmation. When guidance has been developed based on the other sources of information listed in Fig. 1, testing may be necessary to provide confirmatory evidence that (1) the guidance is an acceptable extraction, synthesis, or interpretation of the data, and (2) that the guidance is appropriate to an NPP application.

The great advantage of original research is the ability to focus on the specific design characteristics and human performance issues of interest. It has the disadvantage of being the most costly method of technical basis development relative to the range of issues that can be addressed. Further, such research can be limited in generalizability because any single experiment uses a relatively small sample of operators, a small sample of testbeds (plant types), and may be constrained by the specific way in which HSIs are designed for the study (additional generalizability considerations are discussed in [5]).

Using this guidance development method, draft alarm review guidance was developed using each type of information source listed in Fig. 1 except for original research. Each individual guideline included the technical sources of information that formed its technical basis. This information provides the basis for evaluating the internal validity of guidelines. The technical bases vary for each guideline. Some guidelines are based on technical conclusions from a preponderance of empirical evidence, some on a consensus of existing standards, and others on judgement that a guideline represents good practices based upon the information reviewed. The draft guidelines were then evaluated by independent peer-reviewers who assessed: (1) the internal validity of the guidance, (2) the relevance of the guideline to the nuclear plant setting, and (3) the appropriateness of the guideline for NRC safety reviews. This peer review constitutes the external validation of the guidelines. A revision to the draft guidance based on the reviews was accomplished. The guidance development and technical basis is documented in NUREG/CR-6105 [4] and the guidance itself is integrated into NUREG-0700 [3].

However, there were aspects of advanced alarm system design for which the available information did not fully support guidance development. A program of original research was developed to address these characteristics. The program is discussed in the next section.

III. CURRENT RESEARCH

During guidance development, several human performance issues associated with advanced alarm systems were identified. Those issues associated with alarm processing, availability, and display were considered to have the highest priority. These issues are summarized in Section A below (see [1] for the detailed literature review), the experimental methodology is presented in Section B, and the plan for data analysis is presented in Section C.

A. Processing, Availability, and Display Issues

Alarm Processing: One of the most important objectives in the design of advanced alarm systems is to reduce the number of alarms that occur during plant disturbances. Alarm processing is intended to accomplish this objective. These techniques were developed to identify which alarms are significant and to reduce the crew's need to infer plant conditions. Alarm processing refers to the rules or algorithms that are used to determine the operational importance of alarm conditions. Many of the techniques can be classified into two categories based upon how the information that operators receive is affected. *Nuisance Alarm Processing* techniques essentially eliminate alarms that are irrelevant to the current mode of the plant, e.g., a low temperature alarm on a line that is out of service for maintenance. *Redundant Alarm Processing* techniques analyze alarms to determine which are less important because they provide information that is redundant with other alarms. For example, in causal relationship processing only causes are alarmed and consequences are considered redundant. In addition to reducing the actual number of alarms, however, these redundant alarm processing techniques may adversely affect the information used by the operator for situation assessment, decision-making, or confirmation that the situation represented by the "true" alarm has occurred.

The various processing methods and the degree of alarm reduction should be evaluated for their relative effects on operator performance. However, research that has addressed the effects of alarm processing on performance has been equivocal. Some studies have found an effect of alarm processing on performance while others have not. This could be due to many factors such as type of processing used, degree of alarm reduction achieved, and user familiarization with the system. The effects could also be transient dependent, e.g., dependent on the specific scenario, on the operator's ability to recognize familiar patterns, or on plant type. System complexity should also be considered. The operator, as the system supervisor, should easily comprehend alarm information, how it was processed, and the bounds and limitations of the system. An alarm system combining multiple processing methods may be so complex that it cannot be readily interpreted by operators in time-critical situations. An understanding of this relationship is essential to the development of alarm system improvements and review guidance.

Alarm Availability: This refers to the method by which the results of alarm processing are made available to the operating crew (rather than *how* they are presented, which is alarm display). Two of the techniques that have been used include suppression and dynamic prioritization. Suppression is when less important are suppressed and not presented to the operators, but can be accessed by operator request or by the alarm system based upon changing plant conditions. Dynamic prioritization is when less important alarms are presented to operators but somehow distinguished from those that are more important, such as presenting them in a different color or in a different location than other alarms.

Suppression also removes potentially distracting alarms; however, since they are accessible on auxiliary displays, additional workload may be imposed by requiring operator action to retrieve them. Dynamic prioritization does not conceal any information from operators. However, the operator must perceptually "filter" alarms (e.g., scan for red alarms) and a potential, therefore, exists for distraction from less important alarms. Thus, there are tradeoffs between these approaches and an issue remains concerning when the various options should be employed.

Alarm Display: Alarm displays can be considered along three dimensions: spatial dedication (whether an alarm is always displayed in the same physical location or in variable locations), display permanence (whether an alarmed is always visible or visible only when in an alarmed state), and integration (whether that alarms are presented as a separate system or integrated with other process information. These three dimensions distinguish three main types of alarm displays. Spatially-dedicated continuously-visible (SDCV) alarm displays provide a display of information in a permanent location. Lighted tile alarms are an example. Temporary alarm displays, such as a VDU message list, display alarm messages only when the alarm is in a valid state. Specific alarms usually not appear in spatially dedicated location although they may always be presented on the same VDU. Integrated alarms present alarm information as an integral part of other displays, such as process displays. For example, if alarms are built into a system mimic display, trouble with a component such as a pump can be depicted by a change in color or flashing of the pump icon. These displays may be in a fixed or variable location and are typically not permanent displays. While alarms have traditionally been separate information systems from other indicators, it is thought that the operator's information processing is supported by integration of information into a single displays. The benefits of these types of displays are thought to include: (1) enhancement of parallel processing (lowering cognitive workload), (2) enhancement of the operator's ability to better understand the relationships between display elements, and (3) enhancement of the operator's ability to develop a more rapid and accurate awareness of the situation.

SDCV displays are often preferred by operators and have been shown to have performance advantages under high-alarm conditions. But, placing all alarms on such displays (potentially many thousands of alarms in advanced plants) leads to the alarm overload problem for operators.

VDU message lists have not been completely successful alternatives, however. Message lists have been demonstrated to be problematic in high-alarm conditions. Further, although the research is limited, integrated graphic displays have not been shown to improve performance. To serve the different functions of the alarm system, multiple display formats may be required. Thus the display format and the degree to which alarm information is integrated with other process information are important safety considerations. The role, relative benefits, and design of each type of alarm display format in the presentation of alarm information is an issue.

B. Experimental Methodology

In order to help address these issues, an experiment was performed to evaluate the impact of alarm processing, availability, and display characteristics on plant and operator performance. The extent to which alarm numbers are reduced is a function of the alarm processing techniques that are applied. In this study, a variety of alarm processing methods were employed that are representative of near-term applications, and therefore, near-term regulatory review considerations. Three levels of alarm reduction were used. The first processed nuisance alarms to achieve moderate alarm reduction (called Tier 1 processing). The second processed redundant alarms, which in combination with nuisance alarm processing, achieved maximum reduction (called Tier 2 processing). A third condition of no alarm processing was used to provide a baseline for comparison (called Tier 0 processing).

The differential effect of two types of alarm availability was evaluated: suppression and dynamic prioritization. In the suppression condition, less important alarms were not presented in the primary alarm displays but were available to operators on a suppressed alarm list. In the dynamic prioritization condition, less important alarms were color coded to indicate their status.

Three types of VDU-based primary alarm displays were compared: a dedicated "tile-like" format, a mixed tile and message list format, and a mixed integrated graphic and message list format. The graphic provides alarm information integrated into process display formats. These display formats enabled the examination of two aspects of alarm display design: spatial dedication and degree of integration with process information. A secondary alarm display consisting of a chronological event list was also available to operators in each condition.

The various types of processing, availability, and display were combined to form eight experimental conditions, i.e., unique alarm system configurations (see Table 1). In addition to varying alarm characteristics, two types of scenarios were used: complex and simple. Eight exemplars of each were developed for the study.

The tests were conducted using the Human-Machine Laboratory (HAMMLAB) at the Halden Reactor Project in Norway. The plant model simulates a pressurized water reactor power plant with two parallel feedwater trains, turbines and generators. It is closely related to the plant model used in the large scale training simulator at the Loviisa nuclear power station in Finland. The participants were professional nuclear power plant operators from the Loviisa plant. Six crews of operators participated each made up of a reactor operator and turbine operator. Each crew made 16 experimental trials, two in each of the eight alarm conditions (one with a low complexity scenario and one in a high complexity scenario). There were a total of 16 scenarios so that no scenario was used more than once for each crew. The order of presentation of scenarios was balanced, as was the relationship between individual scenarios and experimental conditions.

Table 1
Experimental Conditions

			Processing			
	P1		P2		P3	
Availability		NA	A1	A2	A1	A2
Display Type						
D1	1		7			
D2	2	3	4	5	6	
D3			8			

Notes:
D = Displays (D1: tile format; D2: tile+message list; D3: integrated+message list)
P = Processing (P1: none; P2: Tier 1-nuisance; P3: Tier 2-redundant)
A = Availability (A1: prioritization; A2: suppression)
Each experimental condition included both levels of complexity (not shown in the table for simplicity).

The measurement of performance in the study included process measures, operator task performance, and operator cognitive processes (e.g., situation awareness and workload). The subjective opinions of the operators were also obtained.

C. Data Analysis Plan

The experimental trials were recently completed and we are currently analyzing the data. The primary objectives of the analyses, and the experimental condition comparisons related to them, are:

1. To determine the effect of spatial dedication on performance; with scenario effects and controlling for processing and availability: Experimental Condition 1 vs 2 and Experimental Condition 7 vs. 4 vs. 8.

2. To determine the effect of alarm integration on performance: Experimental Conditions 8 vs. [4&7].

3. To determine the effect of alarm reduction and processing type (tier) on performance; with scenario effects included, collapsing across availability, and holding display constant: Experimental Conditions 2 vs [3&4] vs [5&6] and Experimental Conditions 1 vs 7.

4. To determine the effect of alarm availability and the interaction of availability and processing on performance; holding display constant: Experimental Condition [3&5] vs [4&6]. (This analysis will also examine the effect of processing with the effects of availability (A) and the interaction of AxP segregated)

5. To determine the effect of the interaction of display type and processing; with scenario effects included: Experimental Conditions 1 vs 2 vs 7 vs 4..

6. To determine the effect of scenario complexity and its interactions with other variables on performance (analyzed in each comparison listed above).

These effects are primarily being tested with repeated measures analyses of variance.

IV. CONCLUSIONS

The nuclear and human factors communities have developed a significant database upon which HFE review guidance for advanced alarm systems was developed. Information supporting guidance development was available not only from alarm guidance documents, but also from published reports of research and operational experience. Further, advanced alarm systems, particularly those utilizing computer-based interfaces, share many HSI characteristics with other control room resources. Thus HFE principles associated with VDUs, graphics displays, dialog structures (such as menus and command language) and computer input devices (such as touch screens, keyboards, and trackballs) are applicable to alarm systems. This information was used to develop HFE guidance for the review of alarm systems.

It was also found that there remain notable human performance issues related to alarm processing, availability, and display. The use of focused research to better understand these issues is contributing to the development of guidance in these areas. The data is currently being analyzed and will be reported on shortly.

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Addressing the Human Factors Issues Associated with Control Room Modifications

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Abstract: Advanced human-system interface (HSI) technology is being integrated into existing nuclear plants as part of plant modifications and upgrades. The result of this trend is that hybrid HSIs are created, i.e., HSIs containing a mixture of conventional (analog) and advanced (digital) technology. The purpose of the present research is to define the potential effects of hybrid HSIs on personnel performance and plant safety and to develop human factors guidance for safety reviews of them where necessary. In support of this objective, human factors issues associated with hybrid HSIs were identified. The issues were evaluated for their potential significance to plant safety, i.e., their human performance concerns have the potential to compromise plant safety. The issues were then prioritized and a subset was selected for design review guidance development.

I. INTRODUCTION

As part of modifications to control systems and human-system interfaces (HSI) in existing nuclear power plants (NPPs), advanced technologies that are predominantly based on digital technologies are being introduced. The result of this evolution is that hybrid HSIs are created; i.e., HSIs containing a mixture of analog and digital technology. While the introduction of advanced HSI technology is generally considered to enhance system performance, there is also the potential to negatively impact human performance, spawn new types of human errors, and reduce human reliability. Two examples of how hybrid HSIs could potentially impact safety follow [1]. In the first example, a keyboard entry coupled with a mispositioned system panel switch led to the lockup of a microprocessor-based overhead, annunciator system that went undetected for over one hour. A subsequent investigation revealed that the annunciator system could be locked up if an operator initiated a specific input twice while the system was connected to the wrong computer port. In another event, an operator assumed manual control of a full-range digital feedwater control system during power ascension, and tried to "bump" open the feedwater valve using a series of short intermittent key presses. However, the operator was unaware that each press corresponded to only about 0.1% demand, and the series of key presses translated into negligible changes in valve position demand. As a result, the plant tripped on low steam generator level. A contributing factor was that the feedback provided by the new digital controller to the incremental manual manipulations was not as clear as the floating needle indication of the former analog system. Additional examples may be found in [2,3]. Thus, it is important to consider the potential effects of these technologies on personnel.

The U.S. Nuclear Regulatory Commission (NRC) is sponsoring research at Brookhaven National Laboratory (BNL) to better define the effects of hybrid HSIs on personnel performance and plant safety and to develop human factors engineering (HFE) guidance to support safety reviews. Should a review of plant modifications involving a safety significant aspect of hybrid HSIs be necessary, such guidance will be needed to provide the NRC staff with the technical basis to help ensure that the modifications do not

The first task was to identify human performance topics and issues related to hybrid HSIs based upon literature, interviews, and site visits. Current and future HSI technology changes were categorized and the potential effects of the changes on personnel performance were examined. Hybrid HSI effects stem from both the new technology itself and its interaction with the analog technology. The effects can be related to: (1) *personnel role* - a change in functions and responsibilities of plant personnel such as may be caused by a change in plant automation; (2) *primary tasks* - a change in the way that personnel perform their primary tasks which are tasks directly involved with operating the plant, such as process monitoring, situation assessment, response planning, and response execution and control; (3) *secondary tasks* - a change in the tasks the operator must perform when interacting with the HSI, such as navigating through displays, searching for data, choosing between multiple ways of accomplishing the same task, and deciding how to configure the interface, but are not directly related to operating the plant; (4) *cognitive factors* - a change in the cognitive factors supporting personnel task performance, such as situation awareness and workload; and (5) *personnel factors* - a change in the required qualifications or training of plant personnel.

The human performance topics included:

- Changes in Automation
- Alarm System Design and Management
- Information Design and Organization
- Display Device Characteristics
- Soft Controls
- Computer-Based Procedures (CBP)
- Computerized Operator Support Systems (COSS)
- Maintenance of Digital Systems
- Configuration Control of Digital Systems
- Staffing and Crew Coordination
- Design Analyses and Evaluation of Hybrid HSIs
- Upgrade Implementation (e.g., transition to new HSIs, personnel acceptance, and training).

Several general human performance issues were identified. There is an overall trend away from spatially dedicated HSIs, which support parallel processing of information, toward virtual work spaces which introduce new demands for serial access to information and controls. This can result in greater cognitive workload and more time spent performing secondary tasks. Computer-based systems can also add to plant complexity and personnel needs for interacting with these complex systems are often inadequately addressed. For example, the systems do not always consider the need for information in the context of the operator's current tasks, goals, and objectives or the need for feedback to the operator from computer systems actions. Having a good mental model or understanding of how computer-based systems work is essential to proper monitoring, supervision, and maintenance of plant systems. Failure to account for the operator's need to supervise plant systems may result in poor situation awareness and a sense of being out-of-the-loop. In addition, personnel concerns, such as training and acceptance, are significant considerations in the introduction of new technology (see [3] for a discussion of the human performance issues).

While numerous specific human performance concerns have been identified, it does not necessarily follow that they are all safety significant. The following is a discussion of the safety evaluation conducted for the human performance topics identified above (see [4] for more detail).

II. METHODOLOGY

The safety evaluation methodology consisted of the following steps: (1) Preliminary Screening, (2) Safety Significance Analysis, (3) Initial Prioritization, (4) Peer Review, and (5) Final Classification and Prioritization. In the first step, topics were screened out if they were already being addressed by the NRC in other projects. The second step was the safety significance analysis. The analysis was based on an adaptation of the approach developed by EPRI in *Guideline on Licensing Digital Upgrades* [5], which was endorsed by the NRC in Generic Letter 95-02 [6]. The general rationale and implementation of the method is discussed below.

Commercial nuclear power plant licensees are permitted to make plant modifications without prior NRC review if the provisions of 10 CFR 50.59 [7] for the determination of an unreviewed safety question (USQ) are satisfied. These provisions state that the licensee can (a) make changes in the facility as described in the Safety Analysis Report (SAR), (b) make changes in the procedures as described in the SAR, and (c) conduct tests or experiments not described in the SAR without NRC review and approval prior to implementation, provided that the proposed change, test, or experiment does not involve a change in the Technical Specifications or involve a USQ. A proposed modification is considered to involve a USQ under the following conditions [see 10 CFR 50.59(a)(2)]: (1) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; (2) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; and (3) if the margin of safety as defined in the basis for any technical specification is reduced. The determination of whether or not a USQ may exist is made by the licensee based on a safety evaluation of the proposed change. The purpose of a 10 CFR 50.59 safety evaluation is *not* to determine whether or not a proposed change is safe. Further, a determination that a proposed change involves a USQ does *not* necessarily mean that the change is unsafe. It means that further NRC review is necessary prior to implementation of the change.

The EPRI guidance focusses on digital upgrade issues and was developed to assist licensees in implementing and licensing digital upgrades using the 10 CFR 50.59 evaluation criteria. The evaluation process may be performed qualitatively. The guidance begins with seven primary questions and a set of supplemental questions to help focus the analysis on important considerations. An answer of "yes" to any of the seven primary questions indicates that a USQ exists. The primary questions are:

1. May the proposed activity increase the probability of occurrence of an accident evaluated previously in the Safety Analysis Report (SAR)?

2. May the proposed activity increase the consequences of an accident evaluated previously in the SAR?

3. May the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR?

4. May the proposed activity increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR?

5. May the proposed activity create the possibility of an accident of a different type than any evaluated previously in the SAR?

6. May the proposed activity create the possibility of a malfunction of equipment important to safety when the malfunction is of a different type than any evaluated previously in the SAR?

7. Does the proposed activity reduce the margin of safety as defined in the basis for any technical specification?

The hybrid HSI human performance topics were considered to be within the context of potential plant modifications that need to be reviewed with respect to their potential as USQs. Thus the EPRI guidance was used as a general model for the development of a safety significance analysis methodology for evaluating the hybrid HSI topics. However, it is important to consider two essential differences between the characterizations of the HSI topics and descriptions of actual plant modifications. First, the information describing the topics is less detailed than a description of an actual plant modification. Hybrid HSI topics are generic in the sense that they are relevant to broad classes of upgrades and the full range of operating NPPs. The description for an actual plant modification would contain detailed information regarding characteristics of the specific upgrade. Second, plant-specific information, such as SAR analyses, plant descriptions, and upgrade implementation plans would be available for an actual upgrade, but is not available for generic characterizations of hybrid HSIs.

With these differences in mind, the analysis process was modified somewhat to better reflect its use as a research tool. Each hybrid topic was described in terms related to a potential modification that could be made to existing NPPs. The example modifications were then evaluated using the EPRI guidance. The wording of the questions was modified slightly. The phrase "proposed activity" was replaced with the phrase "proposed modification." Because the evaluation was based on a characterization of an upgrade, rather than an actual upgrade, evaluations of "likely" or "not likely" were applied to the primary questions rather than the more definitive responses of "yes" or "no," which would be used in the evaluation of an actual plant modification. Associated with the seven primary questions were supplemental questions, which addressed specific characteristics of digital systems. A subset of the supplemental questions that pertained to personnel performance was considered in the evaluation. These supplemental considerations generally addressed (1) failure modes that are caused or aggravated by personnel actions, and (2) failure modes and equipment characteristics that have negative effects on personnel performance.

An additional modification related to the findings. In the evaluation of actual plant modifications, a "yes" response to any of the primary questions results in the identification of a USQ. In our analysis methodology, a "likely" response to any of the primary questions resulted in an identification of the topic as "potentially safety significant." Similar to a USQ, the identification of a topic as a potentially safety significant issue does not mean that the types of plant modifications represented by the topic are necessarily unsafe. It means that its human performance concerns have the potential to compromise plant safety. Therefore, should a review be necessary of plant modifications involving safety significant topics, guidance will be needed by NRC staff with the technical basis to help ensure that the modifications do not compromise safety.

A characterization of each topic was developed to serve as a basis for topic evaluations. Some topics were characterized as typical near-term modifications. These were modifications that could plausibly be performed today at an existing plant, such as the installation of a computer-based procedure system. Other topics were concerned with the process by which designs are developed and implemented. These were described in process-related terms. Each characterization also included a description of the human performance concerns that may be associated with the topic.

Using these characterizations, each topic was evaluated using the seven primary questions and the subset of the supplemental considerations from the EPRI guidance. Then an overall assessment of whether a topic was "potentially safety significant" was made based on the seven primary evaluation questions. The evaluations were performed by four BNL personnel with expertise in the areas of human factors, HSI design, NPP operations, probabilistic risk assessment, and SAR analysis.

In the next step, the topics that were identified as potentially safety significant issues were prioritized to support the effective use of research resources in the development of HFE guidance. The prioritization was based on a subjective analysis of (1) the degree to which the topic addressed the performance of personnel directly involved in the operation of the plant versus personnel involved in supporting roles, and (2) the degree to which the topic addressed HSI components that are primary sources of information and control capabilities for operators.

The forth step was to obtain an independent review of the topic evaluation and prioritization. Five reviewers were selected with special expertise in digital I&C, risk analysis, human factors, human reliability analysis, NPP operations, and operator training. The reviewers were knowledgeable in the area of HSI upgrades and effects on personnel performance and plant safety. They were asked to indicate whether they agreed with the overall BNL assessment of the potential safety significance of each topic and whether they agreed with the prioritization of the topics. Following the peer review, a final prioritization of the topics was developed based on input from the independent review.

III. FINDINGS

All of the topics were found likely to be potentially safety significant. The topics of Alarm System Design and Management and Staffing and Crew Coordination were eliminated during the preliminary screening step because they are already being addressed by the NRC in other projects. The topics generally had broad impact on the types of human actions that are important to the role of personnel in the plant, such as monitoring and detection, situation assessment, response planning, and response execution. Thus the safety significance analysis established a link to plant safety.

Once identified as potentially safety significant issues, the topics were then evaluated to determine their relative priority. The topics were organized into three broad categories: High, Medium, and Low. In establishing the High category it was recognized that plant operators are a last line of defense in the case of an emergency. Thus, the High category included those topics that were considered to have a high potential effect on the ability of operators to respond in the case of an emergency, such as an accident or a major transient that could become an accident. This category includes the HSI components that are primary sources of information and control capability and upon which operators rely to support their primary functions of situation assessment, response planning, and response execution. The high category included: Design Analyses and Evaluation, Upgrade Implementation, Computer-Based Procedures, Information Design and Organization, Soft Controls, and Changes in Automation. The medium category included: Configuration Control of Digital Systems and Maintenance of Digital Systems. The low category included: COSS and Display Device Characteristics.

As indicated above, the results were evaluated by peer reviewers. Based on the results of the review a reprioritization was performed. The reprioritization led to the inclusion of Maintenance of Digital Systems in the high priority group.

IV. CONCLUSION

There are important human performance topics associated with hybrid HSIs that relate to both the technology itself, as well as its design, evaluation, and implementation. Using a safety analysis methodology, it was determined that these topics are potentially safety significant. The topics were then prioritized. To-date, draft guidance has been has been developed for soft controls and computer based procedures, using the guidance development process described in NUREG-0700, Rev. 1. [8] Draft guidance for remaining high priority topics is scheduled to be completed in 1998, followed by peer review of all guidance.

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Results of a Nuclear Power Plant Application of a New Technique for Human Error Analysis (ATHEANA)[1]

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Abstract

A new method to analyze human errors has been demonstrated at a pressurized water reactor (PWR) nuclear power plant. This was the first application of the new method referred to as A Technique for Human Error Analysis (ATHEANA). The main goals of the demonstration were to test the ATHEANA process as described in the frame-of-reference manual and the implementation guideline, test a training package developed for the method, test the hypothesis that plant operators and trainers have significant insight into the error-forcing-contexts (EFCs) that can make unsafe actions (UAs) more likely, and to identify ways to improve the method and its documentation. A set of criteria to evaluate the "success" of the ATHEANA method as used in the demonstration was identified.

A human reliability analysis (HRA) team was formed that consisted of an expert in probabilistic risk assessment (PRA) with some background in HRA (not ATHEANA) and four personnel from the nuclear power plant. Personnel from the plant included two individuals from their PRA staff and two individuals from their training staff. Both individuals from training are currently licensed operators and one of them was a senior reactor operator "on shift" until a few months before the demonstration.

The demonstration was conducted over a 5-month period and was observed by members of the Nuclear Regulatory Commission's ATHEANA development team, who also served as consultants to the HRA team when necessary. Example results of the demonstration to date, including identified human failure events (HFEs), UAs, and EFCs are discussed. Also addressed is how simulator exercises are used in the ATHEANA demonstration project.

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Introduction

Over the past several years the U.S. Nuclear Regulatory Commission (NRC) has sponsored the development of a new method for performing human reliability analyses (HRA). A major impetus for the program was the need for a method that would not only address errors of omission (EOOs), but also errors of commission (EOCs). Both Sandia National Laboratories (SNL) and Brookhaven National Laboratory (BNL) have participated in the development of the new method, referred to as A Technique for Human Error Analysis (ATHEANA). Although several documents have been issued describing the basis and development of ATHEANA [e.g., 1,2,3], two documents currently in draft form will ultimately provide the necessary documentation for applying the method. They include the frame-of-reference manual, which serves as the technical basis document for the method, and the implementation guideline, which provides step-by-step guidance for applying the method. Together, the two documents provide the information needed to identify, characterize, quantify, and integrate into probabilistic risk assessment (PRA) models, potential human failure events (HFEs), unsafe actions (UAs), and their error-forcing contexts (EFCs). HFEs, UAs, and EFCs are defined as follows:

- HFE - A basic event that is modeled in the logic models of a PRA (event and fault trees), and that represents a failure of a function, system, or component that is the result of one or more unsafe actions. An HFE reflects the PRA systems' modeling perspective.

- UA - Actions inappropriately taken, or not taken when needed, by plant personnel that result in a degraded plant safety condition.

- EFC - The situation that arises when particular combinations of performance-shaping factors (PSFs) and plant conditions create an environment in which unsafe actions are more likely to occur.

With the completion of the draft FOR manual in early 1997 and the draft IG in April 1997, along with several "step- throughs" of the process by the development team, it was decided that the method was ready for a third party test. Thus, a demonstration of the method was planned for July 1997.

The main goals of the demonstration were to test the ATHEANA process as described in the FOR manual and the IG, test a training package developed for the method, test the hypothesis that plant operators and trainers have significant insight into the EFCs that can make UAs more likely, and to identify ways to improve the method and its documentation. A set of criteria to evaluate the "success" of the ATHEANA method and the demonstration was identified as follows:

1. FOR manual and IG "Work"
   - Documentation is understandable
   - Process is usable

2. Training is effective
   - Motivates team
   - Facilitates use of the FOR manual and IG
   - Enables leader to direct team
   - Results in plant team applying useful retrospective analysis

3. Process identifies demanding scenarios involving errors of commission
    - Plant operators judge the scenarios to be "demanding"
    - Plant identifies and implements fixes for some scenarios
    - Plant believes that ATHAENA can or will identify important problems

4. Users identify improvements in ATHEANA tools and processes

Seabrook Nuclear Station Demonstration Project

The first demonstration of ATHEANA began at the Seabrook Nuclear Station on July 14, 1997, with a 3-day training session provided by the ATHEANA development team, followed by the beginning of the application of the method. The licensee supported the analysis with two individuals from their PRA staff and two from their training staff. Both individuals from training were currently licensed operators and one of them was a senior reactor operator (SRO) "on shift" until a few months prior to the beginning of the demonstration. A PRA expert, with experience in HRA (not ATHEANA), from SNL served as the team leader for the demonstration. Consulting and documentation support on the application of the method was provided by current members of the ATHEANA development team.

The demonstration was scheduled to proceed over a 20-week period, with most of the actual analysis occurring during six team meetings held at Seabrook. The results and status of the current plan are summarized in Table 1.

Table 1
Current Plan, Results, and Status of Seabrook ATHEANA Demonstration

Current Plan	Results	Status
Step 1: Set priorities among initiating events (IEs) based on criteria such as licensees interest or concerns regarding potential problems, IE frequency, time to CD, etc.	Top three IEs: 1. Medium loss-of-coolant accident (MLOCA) 2. Loss-of-offsite power (LOSP) - station blackout (SBO) 3. Transient (followed by anticipated transient without scram (ATWS) under special conditions)	Completed

Current Plan	Results	Status
Step 2: Using guidance from FOR manual and IG, identify possible HFEs and associated UAs for functions identified in scenarios.	MLOCA (3 critical functions) • Makeup - 6 HFEs identified • Heat removal - 6 HFES identified • Long-term heat removal -5 HFES LOSP - SBO (3 critical functions) • Heat Removal - 5 HFEs identified • Support (Diesel generators (DGs) and/or cooling to DGs) - 5 HFEs • Depressurization (manual) - 6 HFEs identified	Completed for MLOCA and LOSP-SBO scenarios ATWS - Pending
Step 3: Identify/derive potential EFCs that could lead to identified UAs	See discussion of interesting scenarios below.	Partially completed for MLOCA sequences. LOSP-SBO and ATWS - Pending
Step 4: Conduct simulator exercises to evaluate impact of reasonable EFCs on UAs (includes information from debriefing of operators)	--	MLOCA scenario scheduled for week of October 6, 1997. LOSP-SBO scenarios planned for November 1997.
Step 5: Quantifcation of EFCs and HFEs	--	Planned for early November 1997
Step 6: Document demonstration and submit method and results of demonstration for peer review	--	Peer review scheduled tentatively for February 1998

As indicated in Table 1, simulator exercises are an important step in the ATHEANA demonstration. They were used in the ATHEANA demonstration to assess the response of actual operating crews to identified EFCs. The goal was not only to see if certain HFEs actually occurred, but also to be able to discuss with operators their perceptions of the scenario, even if they were successful in responding to the initiating event. Even though some crews may handle the situation, their sense of how the EFCs might effect other crews could be relevant to the quantification of the potential HFEs, given the EFCs. The use of the simulator runs also provides an opportunity see what other aspects of the simulated scenario may affect operator performance and to obtain ideas from the operating crews about nonsimulated EFCs that might influence their performance.

Findings and Conclusions

Process. At this stage of the analysis, it appears that the ATHEANA process is working well. The IG's search process and the FOR manual's guidance tables, theoretical discussions, and summary of operating experience are providing a means for nuclear power plant licensees and others to identify potential HFEs, UAs, and EFCs. Through application of the method, interesting scenarios (described in more detail in next section) are being identified and plant personnel on the HRA team are learning to think about and examine potential human errors in different ways. In applying the method, however, the HRA team found several areas where the process documentation and working tools could be improved. We have found that additional guidance must be provided for the search for EFCs and that the tables developed to document and provide a paper trace of the analysis can be improved. In addition, we have come to realize that the training for application of the method must have later modules for EFC search and quantification. Finally, it was found that while the process can be fairly time-consuming the first time through when the analysts are just beginning to understand the method, the analysis rate improves with practice.

A benefit of the process is that it has helped plant personnel identify opportunities for improvements in plant procedures. Specifically, the MLOCA scenario they developed has led to an unexpected sequence of events that might lead to confusion when using the existing emergency procedures. The EFC development work has suggested ways to improve performance in difficult areas. It appears that one of the trainers will suggest a procedural change to avoid the potential difficulties. In addition, he has decided that inclusion of a version of this scenario in the training sequence next year would be a good idea and is working to that end.

Most Interesting Scenarios.

As presented in Table 1, the ATHEANA process has been used to develop EFCs for the MLOCA and LOSP-SBO initiators/sequences while the EFCs for the ATWS sequence have yet to be developed. As an example of the results from an application of the ATHEANA process, the following information describing the EFCs associated with the MLOCA initiator/sequence is provided.

The MLOCA sequence constructed during the application of the ATHEANA process consists of the initiating event, two HFEs (with one specific UA analyzed for each HFE), and various other "failures" that provide error-forcing contexts. The two HFEs result in loss of makeup and failure of long-term heat removal, and ultimately lead to core damage. The following descriptions provide insight into the sequence as developed by the ATHEANA process.

MLOCA - The medium LOCA initiating event in and of itself may pose a problem to the operators since they do not generally receive training on this size LOCA and thus are less familiar with the postulated plant conditions and their impact on stepping through the emergency response procedures.

HFEs - Inappropriate termination of makeup: The UA analyzed for this HFE is "Operators stop pump." The specific EFC associated with this HFE/UA deals with the failure of wide-range reactor coolant system (RCS) pressure indications in such a manner that the operators may be led to believe that RCS pressure is higher that it actually is. With this potentially misleading information, along with other accurate information, the operators may elect to terminate injection as allowed by specific steps within the emergency response procedures. For this UA to ultimately lead to a functional failure, the operators would have to continue to believe that termination of makeup is the appropriate action.

- Inappropriate depletion of resources: For this HFE, the UA is "Operators operate pump outside design parameters." The specific EFC involves the failure to receive an "Empty" reactor water storage tank alarm. Since the operators are directed by procedure to stop pumps that take suction from the reactor water storage tank upon receipt of an "Empty" alarm, the failure to receive this alarm may delay operator action long enough for the pumps to be damaged due to inadequate suction head requirements. Given the nature of the postulated pump failure, the operator would have little time to correct the effects of the UA.

Other "failures"that will not necessarily lead to core damage but which have the potential to cause the operators problems as they deal with the event include:

A small (3 gpm) steam-generator tube leak,
A turbine-driven emergency feedwater pump trip on overspeed that can be restored, and
A diesel generator failure that is recoverable.

The main effects from these failures are to reduce the operators' cognitive resources as they deal with the accident scenario and cause support staff to respond to and correct failures that by themselves would not lead to core damage.

Evaluation Against Success Criteria. As noted above, at this stage of the Seabrook demonstration project the ATHEANA process seems to be working well. To illustrate how the demonstration is proceeding, Table 2 presents an evaluation of the ATHEANA method and the demonstration against the success criteria. Additional information regarding progress on the demonstration should be available for the presentation coinciding with this paper.

Table 2.
Evaluation of the Demonstration of ATHEANA Against the Success Criteria

Criterion	Evaluation
Do the FOR manual & IG work?	Self-evaluation: While improvements should be made in some of the guidance and support information contained in the FOR manual and IG and in the process documentation tables, the basic search process for HFEs, UAs, and EFCs appears to be working well. While we have not yet reached the quantification stage, it at least appears that the EFCs being identified are reasonable and in principle quantifiable.
Was the training effective?	Utility evaluation: Initial comments (immediately after training) from the Seabrook personnel who participated in the training were very positive about the overall training package. Some important suggestions for improvements included a more extensive initial overview of the method directed at plant management and a brief review of PRA for the benefit of trainers and operators who need a refresher. It was also suggested that detailed training for later steps in the process, such as quantification, be presented just prior to the beginning of that step.
Did the process identify significant scenarios?	
1. Operators judge them as cognitively demanding?	• Operators on the team thought so. Tests on current operating crews are planned for October and November.
2. Did plant changes result?	• Procedure changes are planned and it is clear that the method will allow plant personnel to consider their procedures from a useful new perspective.
3. Plant believes ATHEANA can identify important scenarios	• Will be evaluated after completion of the demonstration

Criterion	Evaluation
Did the users suggest improvements in the ATHEANA process and tools?	Yes. For example, the team indicated that the tables in the FOR manual used to guide the identification of EFCs, could be tied together in a clearer and more systematic way. In addition, they also indicated that some aspects of the search process for EFCs in the IG could be improved by more directly tieing the description of the process to the use of the tables in the FOR manual.

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Radionuclide Transport in the Environment:
A Generic Research Program

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U.S. Nuclear Regulatory Commission

Abstract: The Nuclear Regulatory Commission (NRC) must assure that licensed activities meet all standards and criteria for exposure of the public to radioactive material released by licensed facilities. To achieve this goal Commission staff must anticipate releases of these materials to the environment and evaluate their movement through the environment to man. This is achieved through models based on the latest transport theories and confirmatory data as well as site specific information collected to customize these models to each site. Current models and data are often too simple and conservative to allow realistic assessments of the dose from releases to the environment. The NRC's Office of Nuclear Regulatory Research has developed a generic research program on Radionuclide Transport in the Environment to enhance the staff capability to model environmental releases from licensed activities and calculate doses for comparison to appropriate standards over appropriate periods of time. The general rationale and structure of this research program is discussed in this paper.

Background: The U.S. Nuclear Regulatory Commission is responsible for the protection of public health and safety and the environment from adverse consequences of its licensing actions. The form of this protection is a rigorous review of all aspects of any particular licensing action which may have adverse impacts. Nuclear power reactors are most frequently associated with these responsibilities because of the large amounts of radioactive materials which are contained in the core of an operating reactor. Containment and control of these materials during both operational and accident situations has been the focus of NRC reactor licensing activities. While the probability of a large release of these materials to the environment has been made exceedingly small by applying such concepts as "defense in depth" and redundant safety and control systems, small amounts of radioactive material will eventually reach the environment as gaseous or liquid releases. Other NRC-licensed activities, such as the use of by-product and source materials, disposal of radioactive waste, and decommissioning of contaminated facilities and sites lack the inventories and energy content of an operating reactor but they still involve radioactive materials which can cause harm to the public if released to the environment. The NRC has the responsibility to evaluate potential doses to the public, both short and long term, from radioactive materials released to the environment by these NRC-licensed activities and to compare these doses to standards of acceptable exposure.

The term "performance assessment" is commonly used to describe the process of assessing the potential exposure of the public to radioactive materials from releases of those materials to the environment. In many cases, when the amount of radioactive material released is very small and the environment involved is simple and easily modeled, this evaluation is straight forward and the predicted doses are well within established limits. When the source term (physical inventory and chemical form) becomes large and/or complex and the environment becomes a complex set of hydrogeological systems and geochemical conditions, the performance is no longer simple and easily modeled. In these cases the evaluation is no longer a simple, deterministic calculation of dose. Uncertainties begin to attend every phase of the analysis and the evaluation becomes probabilistic as opposed to deterministic. The uncertain evolution of the system over time now must be represented by a range of potential futures ("realizations") of the environmental systems and the hazardous contaminants. The existing analytical tools, including the models and associated data, are not rigorous enough to realistically address such complex conditions.

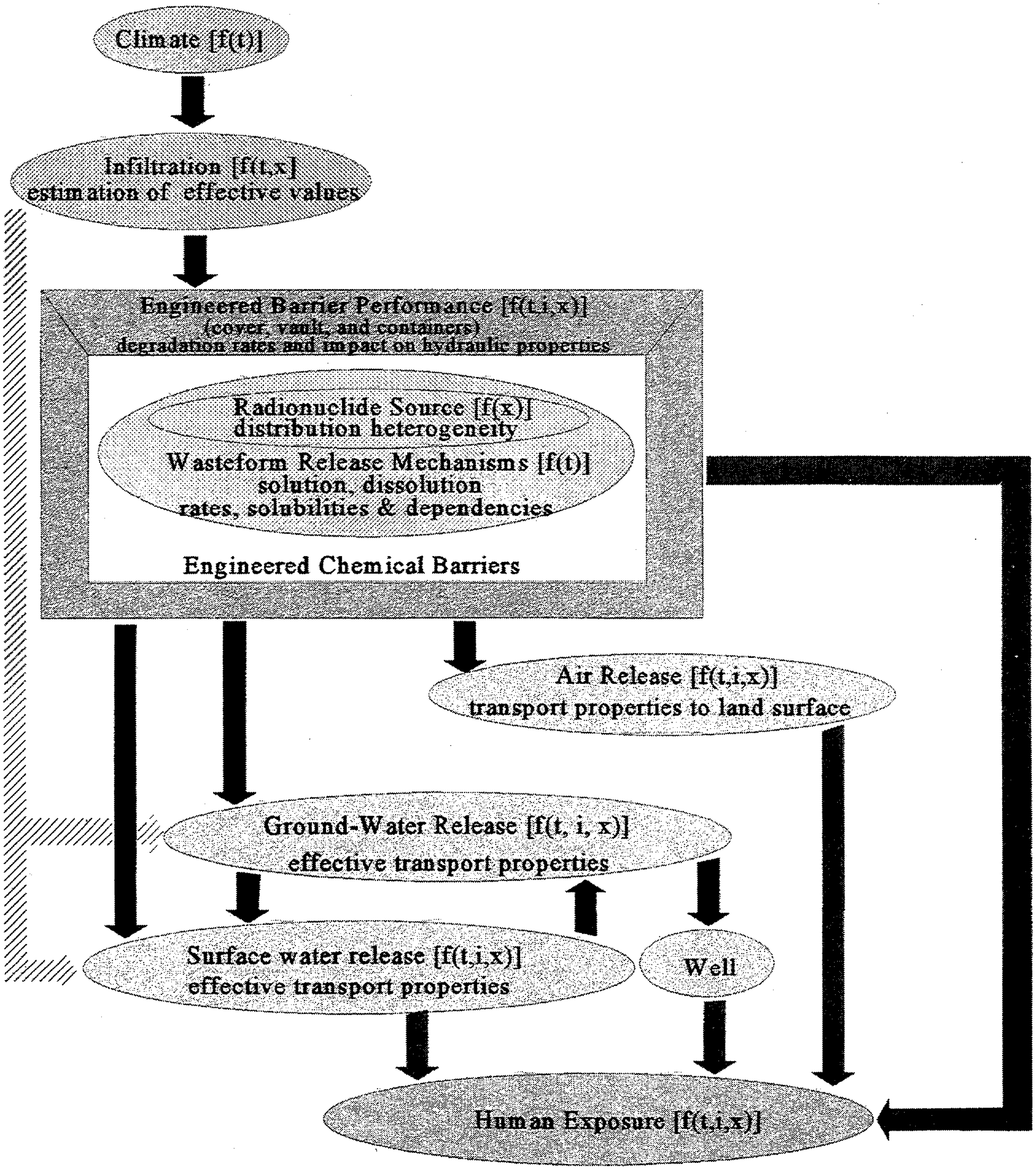
Research Program: The aim of the NRC's Radionuclide Transport in the Environment Research Program is to establish a generic framework of models and supporting data within which to conduct these evaluations and to systematically attack the sources of uncertainty which complicate any performance assessment. If performance assessments are considered to be one subset of a generalized process, it is possible to develop a conceptual model of this assessment process and use this conceptual model to establish areas where research on specific issues can reduce uncertainties in the analytical process. The program is dedicated to evaluating the applicability of existing models and supporting data to various conditions, modifying and refining them where necessary, and developing new models and data for conditions when the existing ones cannot be used.

Generalized Performance Assessment Process: The NRC has done extensive work on performance assessment for high-level (HLW) and low-level waste (LLW) disposal. In addition, the low-level waste disposal work is currently being adapted for use at decommissioning sites. We have examined these three applications of performance assessment to identify ways in which to develop an appropriate generalization or conceptual model. The results of this are shown in Figure 1. This represents a flow diagram for the executive model which will make a single estimate of performance (i.e. calculate a dose) for a given set of conditions (one possible future state of the system). In an actual licensing action, this calculation is repeated many times for different selections of parameter values determined by a statistical sampling of the range and distribution of potential parameter values.

Figure 1 is a simplification of this interacting modeling process. Each box and each pathway may represent one or more conceptual models of a natural or engineered system or waste stream. Nevertheless, organizing our thoughts around this representation of the assessment process helps us question the knowledge and analytical base for each part of the calculation. In the rest of this paper we will follow this strategy to examine where we are in our understanding of these natural processes, and where additional work is needed or planned to help the NRC move toward more realistic assessments of future facility performance. We will begin with the definition of source term, i.e. the origin of our concern because of its potential adverse impact on health. Next we will discuss the disturbed area in the vicinity of the source where various physical and chemical barriers or structures may

Figure 1

Generalized Performance Assessment Process



f(t) : function of time

f(i) : function of infiltration

f(x): function of location

have been used to alter the characteristics of the initial environment. A transport medium is required. Water either from precipitation (rain or snow) or via subsurface flow provides the medium for movement of radionuclides through the soil/rock environment where chemical processes will influence the solubility and sorption of various radionuclide species. Vapors and gases may contribute to an atmospheric pathway for potential exposure. Finally, once radionuclides have reached the environment, both ingestion (drinking water and food chain) and inhalation pathways must be addressed. The overall integration of these analyses into a composite assessment that accounts for uncertainty and statistical variablity is accomplished by the performance assessment strategy which makes parameter selections from pre-defined distributions and evaluates multiple pathways.

Source Term: The portion of the performance assessment least amenable to a generic approach is the source term. It is also potentially a major source of uncertainty. The types of activities for which a performance assessment may need to be carried out include decommissioning of ore processing slag disposal sites, uranium mill tailings disposal sites, low-level waste disposal sites, and high-level waste disposal sites. While no work is being carried out or planned in this area for high-level waste, any of the other areas where a release to the environment is possible are considered fair game for investigation in this generic program.

Low-level waste source term work has been carried out on solidified ion exchange resins including the effects of chelating agents from reactor decontamination, solidified evaporator wastes, and activated metals from reactor components. Microbial degradation of solidified waste forms has also been studied. Work is nearing completion to characterize wastes from full reactor decontamination with the core in place. While these waste streams represent a significant portion of the short half-life activity in LLW and would make a significant contribution to any short term dose predictions in the absence of the stabilization requirements in Part 61, they do not necessarily represent the majority of the long half-life activity and do not make a significant contribution to doses for times beyond 1000 years. Large quantities of long half-life radionuclides are contained in Class A wastes and the in-growth of daughters from the decay of these materials dominates the dose predictions after 1000 years. NRC has done very little work on the LLW source term outside of those few waste streams listed above and this remains as a major source of uncertainty for any LLW performance assessment calculation. However, since most LLW sites are anticipated to be in Agreement States, further LLW source term characterization work has low priority with regard to research funded by NRC license fees.

The NRC's decommissioning program deals with release of formerly licensed sites for restricted or unrestricted use by the public. In most cases the amount of contamination is small, well known, and well contained. The decisions on decontamination and release are reasonably straight forward. The Site Decommissioning Management Program (SDMP) deals with the most difficult sites, i.e. those with larger inventories, greater area of contamination, contamination which may extend into the soil and ground water, or significant contamination associated with migration through complex environments. One set of SDMP sites involves ore-processing slags where radioactive trace components of commercial ores have been concentrated in the slags which are a waste product of the smelting process. These slags are currently the subject of several research studies to characterize the mineralogy, radionuclide content, and long-term stability of the slags as sources of radionuclide contamination. The objective of this work is to develop a range of source terms for the full spectrum of SDMP slag sites, and to provide guidance for the characterization of the contamination at any particular site.

Engineered Barriers: The default option for an environmental transport calculation would assume an undisturbed pathway from the site of the contamination to point of exposure. "Undisturbed" in this case means no direct intervention to enhance containment. In many cases this will not be the case. Reducing doses to acceptable levels will in many cases require some form of direct intervention. This direct intervention can take two forms: contaminants can be exhumed and removed to another location (disposal facility), or engineered systems can be constructed to enhance containment. Some categories of actions such as LLW disposal facilities and uranium mill tailings piles are designed for containment from the beginning. Many older contaminated sites were neither selected for containment potential nor designed to enhance containment. For these sites, relocation or remediation become active considerations if preliminary dose calculations exceed the dose standards applicable to release for unrestricted use.

In the early 1980's NRC sponsored extensive work at Pacific Northwest National Laboratory on uranium recovery disposal sites (uranium mill tailings piles). Until the late 1980's no further work was done on enhancements for near-surface burial. With the passage of the LLW Policy Act which placed the responsibility for developing LLW disposal facilities with the States and started the move toward State Compacts, greater attention began to be placed on the potential benefits of vaults, advanced cover designs and monitoring to assure safe long-term performance of these facilities. NRC efforts in the area of LLW performance assessment consciously began to include engineered enhancements as options. Research was initiated at the National Institute of Standards and Technology to develop models for the long-term containment performance of concrete used in vaults. This theoretical work resulted in the "4SIGHT" code for prediction of concrete performance as a physical barrier over time. Current work is focused on collecting field data to test the predictions of 4SIGHT against existing in-service concretes of various ages and thus provide a sufficient basis for its use in a regulatory setting.

A demonstration project on cover designs was carried out at Beltsville, Maryland. Rip-rap, compacted clay, capillary break, and bioengineered cover performance was compared to the performance of controls with covers of compacted earth and grass. The properly constructed clay and capillary break covers successfully prevented water entry for the life of the project and the bio-engineered covers successfully de-watered the flooded lysimeters and then maintained a soil environment dryer than surrounding soils. This site has been partially decommissioned but several of the covers cells remain as built and continue to be monitored for long-term performance.

The range of engineered barriers which might be applied to constrain the movement of soil contaminants is listed in Table 1. Since new LLW facility designs almost uniformly include some form of concrete vault and engineered cover, these were the most pressing areas to apply NRC resources. However, the 4SIGHT code, which was produced as part of the LLW program, is equally valuable whenever concretes are used in a containment system and are relied on for long term restriction of the movement of radionuclides. Vaults, both above and below ground, have have been proposed for use at decommissioning sites as well as LLW sites. Similarly, stabilized covers have also been proposed for some

TABLE 1
APPLICATIONS FOR ENGINEERED BARRIERS FOR WASTE DISPOSAL

Engineered Barrier Type	LLW	Decommissioning	Uranium Tailings	Remarks
Below Ground Vault	✓	✓	N/A	
Above Ground Vault	✓	✓	N/A	
Earth Mounded Concrete Bunker	✓	✓	N/A	
Disposal Unit Covers -Concrete -Compacted Clay -Multiple layer/capillary break -Bio-engineered†	✓	✓	✓	† Primarily for use as a low-maintenance remediation cover to de-water and prevent additional water passage
Synthetic Membranes	✓	✓	✓*	* May be part of earthen cover
Earthen Cover or Composites (i.e. including synthetic membranes)	✓	✓	✓	
Asphalt Concrete	✓	✓	✓	May be part of earthen cover
Chemical Barriers	✓	✓	✓	
Subsurface Barrier Walls -Soil bentonite slurry -Self hardening slurry -Plastic concrete -Jet grouting -Sheet pilings -Permeation grouting -Geomembranes	N/A	✓	N/A**	** May be applied to contain leakage

contaminated sites. Some of this work on concrete may also assist in assessing the performance of cement slurry walls placed to interdict radionuclide movement at contaminated sites where migration has already been detected. The rapidly evolving field of contaminated site remediation has focused on a number of other techniques for engineering a site to enhance containment. These barriers are also listed Table 1.

Aside from the work already mentioned on concrete and covers, little data exists to support evaluations of the effectiveness of these barriers over long time periods. NRC has plans to continue to pursue the collection of data in this area with long-term performance of soil covers, synthetic membranes and subsurface barriers as the focus following completion of the technical data base for evaluations of concrete degradation.

Transport Processes: We now have a source of radionuclides and some set of engineered enhancements to improve near field containment of those radionuclides. Water and gaseous releases (including water vapor) now become the central focus of the problem. If radionuclides leave the contaminated area and move through the engineered barriers, they will move either as simple dissolved species or as complex chemical species involving organic complexants, microparticulates, or colloids. They may also migrate with soil vapor or as gases in the case of C^{14} and H^3. There are two coupled phenomena involved here which are often treated separately and then combined to yield estimates of radionuclide movement. Infiltration and subsurface flow are assessed using hydrogeologic models which depict the complex three-dimensional geological media and their hydraulic properties through a set of mathematical equations and boundary conditions to determine how water moves through the subsurface environment. Soil chemistry and sorption properties of constituent soil minerals determine how radionuclides move within this flow field. The NRC is only one of many institutions sponsoring work to better understand ground-water flow and transport through geologic media. Extensive work on ground-water flow through low permeability, unsaturated, fractured rock systems was conducted in support of the HLW licensing program. Infiltration and flow through unsaturated and saturated soils and near-surface fractured rock systems was conducted in support of the LLW program. Current efforts are focusing on the uncertainties inherent in choosing the appropriate conceptual models for a given flow system and strategies for monitoring the long-term performance as predicted by performance assessment models for decommissioning sites.

Sorption processes are generally handled through the use of distribution coefficients and retardation factors. This traditional approach develops site specific data on sorption which is then incorporated into the flow models as a constant. This approach is computationally efficient since it is the simplest way to enter sorption into the flow models. However, for the last ten years there has been much criticism in the domestic and international community that this approach does not appropriately consider the variation of chemical conditions such as pH, ionic strength, and the sorbing substrate. Counter arguments have asserted that more realistic models will make performance assessment calculations too cumbersome and inefficient. Geochemistry work sponsored by the NRC since the early 1990's has focused on mechanistic models of sorption processes that will be based on the structure of the sorbing minerals and the way in which radionuclides are bound to this structure. This work has demonstrated that relatively simple surface complexation models can describe the pH and ionic strength dependence of sorption and desorption processes on a specific mineral substrate. This work has so far been extended to only a few radionuclides and sorption substrates. The work is sufficiently promising that a new stage of investigation has been reached in which the surface complexation models will be tested in the context of a performance assessment for a U.S. site. This work is being conducted jointly by the NRC and USGS as one activity under a renewed Memorandum of Understanding between the two agencies. Parallel work is testing the feasibility of incorporating these models into the Sandia Environmental Decision Support System (SEDSS), a multi-purpose, user-friendly code being developed jointly by the NRC, DOE, and EPA to assist in evaluating problems involving any kind of hazardous waste. The increased computational power of new computing systems is anticipated to make a more realistic treatment of sorption possible and thus significantly improve modeling capability.

Pathway Analysis: The NRC has done little work to assess biotic pathways since the early 1980's. A significant amount of work had been completed by the Department of Energy (DOE) and its predecessors, the Energy Research and Development Administration (ERDA) and the Atomic Energy Commission (AEC). In the course of developing a set of test case analyses to support a Branch Technical Position on LLW Performance Assessment, staff assigned to evaluate exposure pathways and identify critical groups determined that much of the information in this area was no longer current and there is a need to initiate work on pathway and critical group analysis. At the present time such work is planned but has not been initiated.

Performance Assessment: NRC began work on performance assessment in the HLW program in the early 1980's. The Sandia National Laboratory was the lead contractor for this work and continued until the work was transferred to the NRC's federally funded research and development contractor for HLW at Southwest Research Institute. Sandia then shifted its focus to LLW and developed a methodology tailored to this type of facility. As NRC work on LLW was phased down, performance assessment efforts were again re-focused to adapt the methodology to decommissioning. In this latest phase the NRC has been cooperatively funding SEDSS development with the DOE and EPA. This performance assessment tool will be versatile enough to attack a wide range of both radioactive and non-radioactive hazardous waste sites. Conversion of the first and simplest version of SEDSS to a user friendly PC version is currently underway. The next step will be conversion to an INTERNET version and then upgrading to include more realistic sub-models to handle the more complex sites.

The approach followed in SEDSS and the earlier NRC performance assessment methodologies and computer codes is to use a statistical sampling approach (Stratified Latin Hypercube Sampling) for all parameters which are most appropriately represented by a distribution of values rather than as a point estimate. The sampling routine selects random values which represent equal probability intervals for successive iterations (realizations) of the computation and then generates a distribution of predicted doses for the modeled system. This approach allows the analyst to define a central estimate and to select low probability outliers for further analysis, i.e. to examine anomalous high or low dose predictions to determine what variables or combinations of variables caused the deviation from the central estimate.

Conclusion: NRC regulates many facilities and activities which have the potential to release radioactive material to the environment. In reviewing licensing actions, the consequences of these actions must be assessed against appropriate criteria and standards. In most cases this requires a calculation of dose to the public over the period for which the potential dose remains significant. Analytical techniques and computational tools have been developed to carry out these calculations. At the present time these analytical tools rely heavily on simplifying assumptions and "conservative" estimates. For some analyses the number of parameters requiring input values for a specific calculation can be in excess of 400 parameters. The cumulative effect of simplifying assumptions and conservative estimates in a computation this complex is not necessarily conservative. A real need exists to address the uncertainties in these evaluations and develop more realistic tools for modeling facility performance. The Generic Radionuclide Transport Research Program, building heavily on existing models and associated data developed previously under the LLW and HLW research programs is systematically attempting to reduce these uncertainties and provide more realistic models for use in the review of these facilities.

REGULATORY FRAMEWORK FOR FINANCIAL ASPECTS OF DECOMMISSIONING NUCLEAR POWER REACTORS

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ABSTRACT

Deregulation of the electric power industry has redefined the nature and role of the electric utility. This has created a situation where the existing financial assurance regulations may no longer be commensurate with the financial arrangements of reactor licensees. Further, when the Commission first promulgated the financial portions of its decommissioning regulations in 1988, it did not explicitly consider that significantly reduced potential risk of reactors in a defueled but not yet decommissioned state. Today's discussion will focus on the financial aspects of decommissioning nuclear power reactors.

I. BACKGROUND

There are three parallel efforts in progress at this time. In the first one, the NRC is proposing to amend its regulations relating to financial assurance requirements for the decommissioning of nuclear power plants. Potential deregulation of the power generating industry has created uncertainty with respect to whether current NRC regulations concerning decommissioning funds and the financial mechanisms will require a modification to account for utility restructuring not contemplated when current financial assurance requirements were promulgated. In parallel, the NRC is proposing to amend its regulations to allow licensees the option of submitting site-specific decommissioning cost estimates to justify decommissioning funding levels below the generic values currently codified as minimum levels. The third activity relates to amending the financial protection requirements and reducing the level of insurance coverage commensurate with the risk reduction after the appropriate spent fuel cooling period following permanent shutdown (PSD) of the reactor. Current regulations do not address the various spent fuel configurations for PSD reactors.

II. DECOMMISSIONING COSTS REQUIREMENTS

Under Section 161 of the Atomic Energy Act of 1954, as amended, the NRC has general authority to regulate the decommissioning of the nuclear facilities and materials that it licenses. When the NRC promulgated its decommissioning regulations in 1988, the agency had determined that decommissioning funding assurance requirements were necessary to assure that the bulk of the funds needed for decommissioning were available when needed to protect public health and safety. Under one of the options, the NRC requires its power reactor licensees to periodically set aside funds in external trust fund accounts in order to accumulate an amount at least equal to the amount specified by the formula in 10 CFR 50.75 after adjusting for inflation. The amounts stated in the regulations are $105 million (1986 dollars) for a full-size pressurize water reactor (PWR) and $135 million (1986 dollars) for a full-size boiling water reactor (BWR). To account for inflation, the rule provides a formula weighted to account for changes in labor, energy, and low-level waste burial costs since 1986. Although, licensees are required to update the certification amount annually using this formula, these updates do not have to be submitted to the NRC. The current regulations do not require licensees to take any immediate action based on these updates.

A reactor licensee may prepare and use a site-specific cost study as long as the estimated decommissioning cost exceeds the combined total of the initial formula amount and the cumulative inflation adjustments as specified in the rule. Many power reactor licensees have chosen to prepare site-specific studies and collect funds toward the amounts identified in the site-specific studies. In part, because the amounts specified in the regulations were generic values based on a typical PWR or BWR site and not intended to precisely estimate decommissioning costs. Some licensees have chosen this approach because they will have to contend with costs that the NRC does not include in its definition of decommissioning. Under 10 CFR 50.2, decommission means "to remove a facility or site safely from service and reduce residual radioactivity to a level that permits: (1) release of the property for unrestricted use and termination of license; or (2) Release of the property under restricted conditions and termination of the license." The NRC does not include in its definition of decommissioning the demolition and removal of non radioactive structures and components and restoration of the reactor site. Also, the NRC excludes from decommissioning the costs of management and the storage of spent fuel.

The current regulations state that a reactor licensee may conduct a site-specific decommissioning cost estimate at any point in a reactor's operating life, but it <u>must</u> do so about five years before the expected termination of operations. This cost estimate will form the basis for collecting any shortfall of funds or disposing of any excess funds derived from the generic NRC formulas. Any shortfalls are to be collected over the remaining five years of facility life. The current regulations require that within two years following permanent cessation of operation, the licensee has to submit a site-specific cost estimate for decommissioning the facility, a description of activities, a schedule of completing the decommissioning, etc.

A number of changes have occurred in decommissioning technology and in the availability and cost of low-level waste disposal. Because of these changes, licensees have made significant efforts to reduce projected decommissioning waste volume. As a result of these efforts and developments in decommissioning technology, the NRC asked PNNL to reassess the cost estimates for decommissioning the reference PWR and PWR plants. These latest PNNL cost estimates are considerably lower than most site-specific licensee decommissioning cost estimates, and even lower still than the minimum decommissioning cost values provided in 10 CFR 50.75(c). The NRC at the present time is collecting data from actual site-specific decommissioning activities and will compare with the PNNL estimates. Therefore, rather than change the values in 10 CFR 50.75(c), the NRC is proposing amendments to allow applicants and licensees to submit a site-specific estimate, which may be lower than the generic values provided in 10 CFR 50.75(c). This would provide greater flexibility in dealing with site-specific issues such as differences in decommissioning methodology, expected waste volumes, and anticipated labor efforts to perform specific tasks. This has the potential to provide a significant reduction of burden to some licensees.

III. FINANCIAL ASSURANCE REQUIREMENTS

Under the existing framework of the financial assurance requirements for decommissioning, licensees are given options for providing the funds necessary for decommissioning. Licensees are also required to revise annually their estimates of the total amount of funds needed for decommissioning. The regulations provide NRC the right to share responsibility for decommissioning funding with rate regulators, such as State Public Utility Corporations (PUCs) and the Federal Energy Regulatory Commission (FERC). Traditionally, the NRC has relied on FERC and PUCs for decisions such as the sources of decommissioning funds (whether rate-payers or licensee stockholders), the timing of funds collections, and the investment in trust funds. This practice is consistent with earlier NRC determinations that traditional cost-of-service rate regulation provides reasonable assurance of funds for operations and decommissioning.

Current regulations allow only those licensees that meet the NRC's definition of "electric utility" to use the external "sinking fund" method of decommissioning funding assurance. The NRC is concerned that given the deregulation and restructuring of the electric utility industry, the possibility exists that new entities may be created that, while fitting the existing definition of "electric utility" would no longer be under the authority of rate regulators.

The NRC also has explicit requirements in 10 CFR 50.82 concerning the release of decommissioning funds from trust accounts. The NRC has regulatory authority to stop any unwarranted withdrawals and to require reimbursement of the trust fund for unwarranted withdrawals already made. Based on the broad authority given under the Atomic Energy Act of 1954, as amended, the NRC also could order trust fund disbursements for a particular decommissioning-related activity, based on the presence of a threat to public health and safety if the activity did not occur.

In response to the anticipated rate deregulation of the electric utility industry, a proposed rule on "Financial Assurance Requirements for Decommissioning Nuclear Power Reactors" was published in the Federal Register on September 10, 1997. Changes in the proposed rule include:

- Revision of the definition of "electric utility" to reflect changes caused by restructuring within the electric utility industry.

- A reporting requirement for nuclear power reactor licensees to provide the NRC information on the status of their decommissioning funding and any changes to their trust agreements for each power reactor at least once every 2 years. However, once the reactor is within 5 years of the projected end of operation, the licensee must submit a report annually.

- Allowing nuclear power reactor licensees to take a 2 percent real rate of return credit on earnings for prepaid decommissioning trust funds and external sinking funds from the time the funds are set aside through the end of the decommissioning period.

IV. FINANCIAL PROTECTION REQUIREMENTS FOR PERMANENTLY SHUTDOWN PLANTS

The current regulations governing insurance coverage for nuclear power reactors are contained in 10 CFR 50.54(w) and 10 CFR 140.11. Section 50.54(w) provides insurance coverage requirements for onsite damage in the event of an accident at the license's reactor, and Section 140.11 provides insurance coverage requirements to protect against offsite liability. These regulations were developed for operating reactors and as such do not recognize the various reactor or spent fuel configurations or take into consideration the reduced risk for accidents associated with permanently shutdown reactors. Currently, licensees are using the exemption process allowed under the regulations to reduce the insurance coverage for these PSD plants. The NRC is proposing to amend its regulations to address these issues in a uniform and consistent manner. This is also part of the NRC's effort to eliminate unnecessary regulatory burdens for power reactor facilities that are permanently shutdown and in the process of decommissioning.

Under 10 CFR 50.54(w), power reactor licensees must obtain insurance coverage from private sources to provide protection against onsite damage in the event of an accident. These monies would allow the licensee to stabilize and decontaminate the reactor and reactor site in the event of an accident. The minimum amount of insurance coverage is the lesser of $1.06 billion or the maximum amount of insurance generally available from private sources.

The regulations in 10 CFR 140.11 require that licensees with facilities designed to produce substantial amounts of electricity (a rated capacity of 100,000 KWe or more) must have and maintain a primary insurance coverage of $200 million from private sources to protect against offsite liability. In addition, licensees must maintain secondary financial protection in the form of private liability insurance available under an industry retrospective rating plan. The current maximum obligation for secondary financial protection for a licensee in this plan is $75.5 million with respect to any nuclear incident. Thus, the total financial protection for offsite liability for any incident would be the primary layer of $200 million, plus the secondary layer of $75.5 million multiplied by the number of licensed power reactors with a rated capacity of 100,000 KWe or higher.

The NRC is proposing to amend its regulations to allow nuclear reactor licensees to reduce onsite and offsite liability coverage during permanent shutdown of the reactors if they meet specified reactor configurations. These reactor configurations will include fresh spent fuel moved from the reactor vessel to the spent fuel pool to a configuration when all the spent fuel has been moved from the reactor site and no other radioactive material is left on site. The proposed amendment would reduce the level of insurance coverage commensurate with the risk reduction after the appropriate spent fuel cooling period following permanent shutdown of the reactor. The proposed amendments would adjust the onsite insurance coverage requirements and the offsite requirements for permanently shutdown reactors based on accidents involving a loss of spent fuel water and on the amount of onsite radioactive inventory such as liquid radwaste in post shutdown modes.

For onsite insurance coverage, 10 CFR 50.54(w) will be amended to reduce the minimum insurance amount from $1.06 billion to $50 million for a condition when spent fuel could tolerate a complete loss of water in the spent fuel pool and the decay heat is low enough to preclude rapid zircaloy oxidation and to $25 million when there is no fuel in the spent fuel pool and no significant source of mobile radioactive material on site.

In parallel, offsite liability requirements as specified in 10 CFR 140.11 would also be adjusted to allow licensees to lower their primary insurance coverage and to be able to withdraw from the industry retrospective rating plan, based on satisfying the requirements specified for several different spent fuel configurations during permanent shutdown. These insurance amounts will be reduced from $200 million to $100 million for the configuration, when the fuel could tolerate a complete loss of water in the spent fuel pool. This amount is based on the potential for significant judgments or settlements resulting from litigation despite negligible offsite consequences. This amount will further be reduced to $25 million, when there is no fuel in the spent fuel pool and no significant source of mobile radioactive material is present onsite.

V. CONCLUSIONS

In conclusion, the NRC is evaluating its regulations related to decommissioning of nuclear power plants. These regulations include adequacy of decommissioning funds as well as financial assurance that these funds will be available when needed to decommission the nuclear plant. The regulations must provide for adequate protection for the public health and safety in the face of changing environment and deregulation of the electric power industry. The proposed amendments of the financial protection requirements for permanently shutdown plants will recognize the reduced risk of accidents after appropriate cooling of the spent fuel and the potential for burden reduction for some licensees.

Implementation of the New Decommissioning Standard

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The U.S. Nuclear Regulatory Commission published the final rule on radiological criteria for license termination on July 21, 1997. This final rule amends 10 CFR 20, Subpart E, and establishes criteria for the remediation of contaminated sites or facilities that will allow their release for future use with and without restrictions. As part of the work associated with implementing the final rule, a decision methodology has been developed to support implementation of the dose assessment requirements in the new Subpart E. A logical, consistent decision process is viewed as a useful tool that will support licensee planning of decommissioning activities and NRC review of license termination requests.

1. Introduction

The U.S. Nuclear Regulatory Commission published the final rule on radiological criteria for license termination on July 21, 1997. This final rule amends 10 CFR 20, Subpart E, and establishes criteria for the remediation of contaminated sites or facilities that will allow their release for future use with and without restrictions. Guidance is being developed in several areas to support implementation of the rule. These areas include site surveys, institutional controls, ALARA, public participation, and dose assessment.

As part of the work associated with implementing the final rule, a decision methodology has been developed to support implementation of the dose assessment requirements in the new Subpart E (see Figure 1). The decision process supports assessment of the entire range of dose modeling options from which a licensee may choose, from changing a single parameter to changing multiple parameters and modifying pathways or models. A logical, consistent decision process is viewed as a useful tool that will support licensee planning of decommissioning activities and NRC review of license termination requests.

Generic exposure scenarios and pathways have been defined based on the NUREG/CR-5512 methodology and can be used without further analysis or justification by licensees who are applying the default scenarios and parameters using the DandD software. The default screening scenarios and pathways provide the licensee with a simple method to demonstrate compliance using little or no site-specific information. The generic models and default parameters are intended to estimate the upper range of the dose that the average member of the critical group could receive. The default parameters were developed probabilistically to control the regulatory risk associated with releasing a site based on source term data alone.

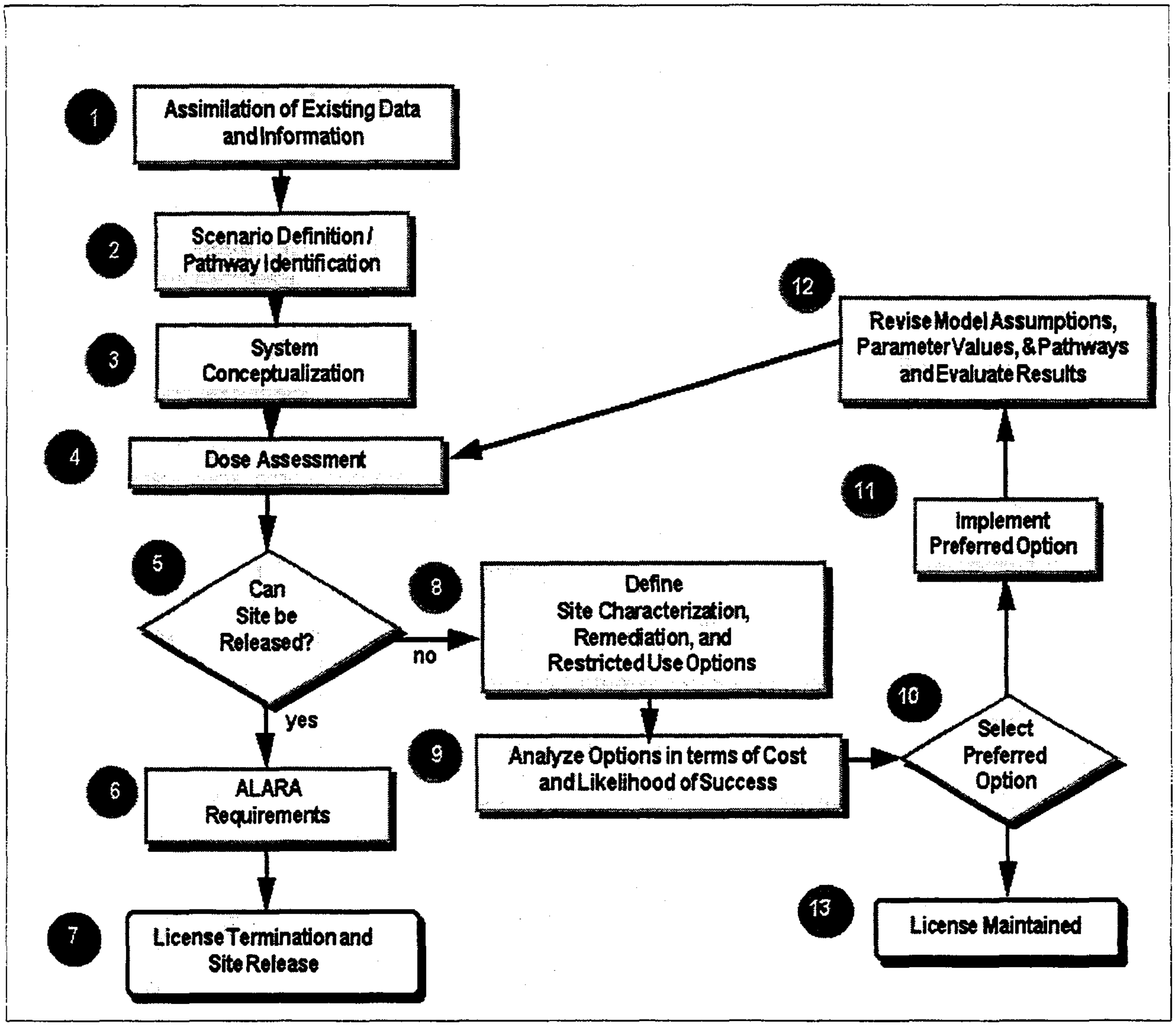

Figure 1 Decision Framework

For licensees with more complex decommissioning situations, the decision process supports the modification of model parameters to allow site specific factors to be taken into account while still using the default models. This allows a licensee to use site-specific values in place of some or all of the default parameters. Thus, the dose estimates are more realistic, but should still be conservative for a particular site based on the use of the default models. The site specific data are used to support modifying or eliminating a particular scenario or pathway, or to demonstrate that a parameter or group of parameters can be better represented by site specific values. Alternative exposure scenarios may be appropriate based on site-specific factors that affect the likelihood and extent of potential future exposure to residual radioactivity.

2. Description of the Decision Framework - Example applications

The purpose of the framework is to provide a logical structure for regulatory decision making within the context of the requirements of the rule on radiological criteria for decommissioning. A useful way to describe the framework is to discuss example applications. Two example applications will be described below: a simple case where detailed modeling is unnecessary, and a more complex case where a decision must be made between unrestricted and restricted release.

2.1 Case 1 - Little or no contamination

The first step of the decision process involves gathering and evaluating existing data and information. The licensee would check their records to determine the types and amounts of radioactive material they possessed on their site. They would also gather information about any surveys and leak tests that had been performed, as well as any records that would support their ability to "Certify the disposition of all licensed material, including accumulated wastes, by submitting a completed NRC Form 314 or equivalent information" [10 CFR 30.36(j)(1)]. In this example, the licensee determines that all waste has been properly disposed, sources have been properly transferred to another licensee, and minor amounts of contamination have been detected inside a laboratory building during routine surveys.

The second step in the decision process involves defining the scenarios and pathways that are important for the site dose assessment. For a simple case, this step has already been completed by the NRC, based on the generic scenarios and pathways for screening that have been defined and described in NUREG/CR-5512, Volume 1. For the example licensee in Case 1, the building occupancy scenario applies, with the associated default inhalation, secondary ingestion, and external exposure pathways. Building occupancy applies to situations where contamination exists on interior building surfaces and where the building will be re-used for commercial (not residential) purposes following license termination.

Step three of the decision process involves system conceptualization, which includes conceptual and mathematical model development and assessment of parameter uncertainty. For the simple example of case 1, this step has already been completed by NRC, using the models described in NUREG/CR-5512, Volume 1, and implemented in the DandD software. The licensee in this example would use the DandD software with the pre-existing scenario definitions and default parameters.

Step four involves the dose assessment for the site, which in this example case involves running DandD with the maximum surface contamination concentration information from the existing building surveys. The maximum survey results are used because, if the dose assessment using these values indicates that the dose is below the 25 mrem/yr criterion, and assuming the surveys meet minimum standards, there will be a high assurance that the site meets the dose requirements and additional refinement of the source term will be unnecessary. Step five is then simply answering the question of whether the dose assessment results from the model are less than the dose criterion of 25 mrem/yr in 10 CFR 20, Subpart E. In this example, the model results are much less than the 25 mrem criterion.

Based on the results in step five, the licensee can proceed to step six and satisfy any remaining ALARA requirements. With the ALARA requirements satisfied, the licensee would complete the paperwork requirements in step seven, including documenting the survey results used to calculate the source term and the model output, and would have their license terminated by the NRC.

2.2 Case 2 - Complex Issues

For the purposes of this example, the licensee is interested in terminating the license for an outdoor location that is believed to have areas of soil contamination from leaks in a waste tank. Although this licensee has a more complex situation than that described in Case 1, they would still follow the same steps described above, at least for the first iteration. As before in step one, they would gather as much information as possible about their site, including radionuclides and processes used, quantities and forms of material that might still remain on site, and anything else that would be useful for performing a site dose assessment.

For the scenario definition and pathway identification in step two, the licensee in this example decides to begin the decision process by using the pre-defined scenarios and pathways in the residential scenario (soil contamination) described in NUREG/CR-5512, Volume 1. In step three, they also accept the default parameters and use the DandD software. For the step four dose assessment, they run DandD using a source term developed from the information gathered in step one, and which is the maximum reasonable value they believe they can defend.

Based on the results of step four, in step five it is clear that the site does not meet the Subpart E dose criterion of 25 mrem/yr. The licensee would therefore proceed to step eight and begin defining their options. Note that there are basically three options that the licensee could apply either alone or in combination: Option 1 - Activities that reduce uncertainty (information/data collection), Option 2 - Activities that reduce contamination (remediation), and Option 3 - Activities that reduce exposure (land-use restrictions). Table 2.2.1 lists some of the options that a license could consider, including two related to reduction of uncertainty, one related to reducing contamination, and one related to reducing exposure.

Only a limited number of sites will need to perform complex dose assessment and options analyses, with most sites performing an options analysis that is relatively simple and straightforward. For example, a site with a small, contained source of contamination that is obviously simple to remove would not perform extensive analyses on large suites of alternative data collection and remediation options. The same may be true for certain complex sites, where the configuration of the contamination, site conditions, or regulatory requirements cause the options for proceeding forward to be relatively limited and straightforward. The sites which will benefit the most from this options analysis are those with complex contamination situations where this process can be used to analyze a variety of simple and complex options and define the most effective and cost-efficient decontamination and decommissioning strategy.

For the first option, activities that reduce uncertainty, it is useful to begin by looking at the default parameter values in the NUREG/CR-5512 model and what they represent. The default parameter values for the NUREG/CR-5512 modeling (that have been implemented in DandD) were developed based on probability distributions representing the expected variability across all NRC sites in the country. A probabilistic parameter analysis was performed to select a set of default parameters that meet the NRC's requirements to control the regulatory risk associated with releasing a site based only on source term information. The regulatory risk is defined as the risk that a site will be released when it exceeds the dose criterion. The risk is controlled by selecting screening parameters that, as a set and within the context of the specified model, provide a specified level of confidence in the dose estimate and control the amount by which the dose could exceed the criterion. The parameter analysis also provided information regarding the valid ranges for site specific parameter changes that a license could propose without an additional uncertainty analysis. As a consequence, the licensee needs little supporting information to defend changes to the parameter values that are within the limits specified in the parameter analysis. This is important in evaluating the relative worth of collecting additional data on these parameters under Step 9 of the decision framework.

For example, the probability distribution used in defining the default values for radionuclide sorption in soils for the NUREG/CR-5512 residential scenario models is based on the variability across all possible soil types at NRC licensed sites. To provide the NRC with an acceptable level of regulatory risk in terminating the license for a site based only on residual contamination data, the default value for sorption coefficient defined in the preliminary parameter analysis is representative of a very clean sand. Therefore, in this step of the options analysis many sites would be able to propose that the sorption be increased significantly. The associated cost for this activity could, for example, be the cost of obtaining a soil conservation map (covered under Step 9). This approach of moving away from the "prudently conservative" values used in the NUREG/CR-5512 modeling based on site-specific information could be used by all sites until the point that further reduction in simulated dose would require model changes. At that point, probability distributions for the new model parameters would have to be developed and defended by the licensee.

As stated above, the options that have been identified in this iteration include two related to reduction of uncertainty, one related to reduction of contamination, and one related to reduction of exposure. The first option would reduce uncertainty in the source term, while the second would replace the default kd with a more site specific value based on the site soil type. The third option listed in Table 2.2.1 would result in an actual reduction of the quantity of residual radioactivity remaining on the site. If the final option, reduction of exposure, were pursued, the licensee would be required by 10 CFR 20, Subpart E, to demonstrate that unrestricted release was not ALARA. This would require additional site specific modeling to ensure that the decision has a sufficient basis.

Table 2.2.1 - Example Options Definition Table

Expectation	Effect on Dose	Action
Source is believed to be lower concentration than currently modeled	Simulated dose expected to decrease as concentrations decrease	Collect field data to better characterize source distribution
Soil type is expected to be predominantly clay and consequently have higher Kds	Simulated dose expected to decrease as availability of radionuclides to the receptor is decreased	Collect literature and soil map data to defend alternative soil type/texture
Enough soil is expected to be permanently removed to decrease source concentrations so dose level is acceptable	Actual available mass of contaminant decreases, hence simulated dose would decrease	Remediation by soil removal
Controls are expected to remain in place for the duration of the compliance period (if controls fail, simulated doses are between 25 mrem and 100 mrem)	Restrictions will limit uses for site while controls are in place to limit exposure time and pathways to individual; simulated dose will decrease	Set land use restrictions and apply for restricted release

The licensee now moves to step 9, analysis of options in terms of cost and the likelihood of success. To evaluate the likelihood of success, an analysis of the potential outcome (consequence analysis) will need to be performed for each of the options. Depending on the option, this consequence analysis could be very simple (e.g., the option is complete remediation and the consequence is effectively restoring the system to an acceptable condition) to as complicated as refining and expanding the dose assessment. The cost and time required to complete each option must also be estimated. The consequence analysis should also address the uncertainty associated with each potential outcome. The desired endpoint is a determination of the likelihood or probability that employing a given option will result in meeting the criteria of 10 CFR 20, Subpart E.

The result of the activities performed under Step 9 is a logically organized list of options, and the corresponding cost, likelihood of site release (probability of success), and other important considerations given that the option is pursued. Table 2.2.2 contains examples of how the options could be organized. In some cases, the decision regarding the preferred option will be obvious; for example, a low cost of success and failure, high probability of success option will always be selected over a high cost, low probability of success option. However, the preferred option will not always be obvious, and additional analysis may be required for sites attempting to balance complex issues.

Table 2.2.2 - Example Options Analysis Table

Alternative Action	Cost (if successful)	Cost (if unsuccessful)	Probability of Success	Required Outcome*
Collect field data to better characterize source distribution	$$	$$	medium	dose less than 25 mrem
Collect literature data to defend alternative soil type/texture	$	$	medium	dose less than 25 mrem
Remediation by soil removal	$$$	$$$	high	dose less than 25 mrem
Set land use restrictions and apply for restricted release				dose w/ controls less than 25 mrem; dose w/o controls less than 100 mrem

*These assume each option is performed in isolation. If performed in combination with other options, each option on its own would not need to achieve a dose less than 25 mrem

To analyze the potential outcome of the selected options, the licensee can use the DandD software to perform some low cost "what-if" calculations. For example, they can review the existing information about their source term and try to estimate how it would change based on additional characterization. Based on the quality of the existing information, they may be able to modify the source term and obtain a less bounding value. This modified source term would then be input into the model and a revised dose estimate calculated.

In the same way, the licensee could review site specific or regional data to determine the predominant soil type at their site. If the soil type is not well characterized by a clean sand, as was used to define the default soil parameters, the licensee could investigate the impact of changing parameters associated with soil type, such as kd. This process can be continued for other model parameters that the licensee believes could be changed based on site-specific information. This is similar to performing an informal sensitivity analysis, and will help focus attention to those parameters likely to have the most impact on the calculation of dose. The licensee can then direct resources to reducing the uncertainty in those parameters, or can determine that a different approach is necessary before any higher cost activities, such as soil removal or site surveys, are begun.

For this example case, it is assumed that a preliminary evaluation of the remediation option indicates that it is not cost effective to remove the contaminated soil and transport it off site. However, the preliminary analysis is based on the default dose screening and initial bounding estimate of the source term, both of which impact the estimated soil volume requiring remediation, and the cost of remediation. These estimates will change as more site-specific data is obtained, which may make remediation a more reasonable option at another point in the decision process. At this point in the decision process, the idea is not to permanently eliminate options from further consideration, but rather to select the optimum approach for the current state of knowledge.

As noted above, use of the option of setting land use restrictions requires a demonstration to the NRC that further reduction in dose levels to unrestricted use is not ALARA. Thus, the absence of cost and probability of success values for the restricted release option is used here to illustrate two important points. First, given the NRC's preference for unrestricted release, licensees are expected to fully evaluate unrestricted release for the first iteration through the decision process. Second, any information gathered to support other options can be used in a later iteration to support restricted release if necessary. It should be noted that the dose modeling must include as much site-specific information as necessary to provide a reasonable evaluation of future impacts, both with and without institutional controls in effect, to show compliance with restricted release criteria. The regulatory activities that will need to be completed prior to NRC granting a license termination under 20.1403 include (1) development of a safety evaluation report, (2) an environmental assessment (and, possibly, an environmental impacts statement), and (3) possibly, additional requests for information to allow staff review of the license amendment. In addition to the safety analysis report for the license amendment, licensees requesting license termination under restricted release will need to submit an environmental report, which will include the cost-benefit analysis for the ALARA determination. Under 20.1403, licensees would also need to seek advice from affected parties in the community regarding the restrictions on use.

This step in the decision framework should support an evaluation of the cost and time impacts of both success and failure. Generally, low cost / high likelihood of success options, or combinations of options, are preferred. This step should also include ALARA considerations, in terms of cost/benefit calculations as well as qualitative considerations. With regard to costs, the licensee should consider that if the option(s) selected are successful, the license will be released and further costs will be minimized. However, if the selected option(s) are unsuccessful, it may be necessary to perform additional characterization or remediation, or there may need to be an evaluation of restricted use (with its associated costs).

Once the various options have been evaluated, the preferred option can be selected in step 10. Based on the DandD analysis and cost estimates for this example, the licensee decides to perform additional characterization of the source term, with the expectation that this will result in the source term estimate being reduced. The additional characterization will also involve obtaining data on the site soil type to support revision of the default kd. The combination of these two options should have a medium cost and a high likelihood of success. At this stage in the analysis, unrestricted release is preferred, and therefore restricted release not considered further at this time.

Under step 11, the preferred option is implemented. The licensee develops a characterization plan that will support both radiological and soil data requirements, then obtains regional soil maps and performs a radiological site survey. If the licensee has a very high expectation that the additional information will be sufficient to support a revised dose assessment that is less than or equal to 25 mrem, it may be worthwhile to design the site survey so that it can be used as a final site survey. That is how the licensee proceeds in this example. However, it is important to note that the final site survey has more extensive requirements than may be needed if the site requires remediation. The extra cost of a final site survey must be weighed against the need to repeat the survey at a later time.

Once the preferred option has been implemented, the model assumptions, parameter values, and pathways (as appropriate) are revised in step 12 of the decision process. For this example, parameter values associated with soil type (kd) and source term are modified based on the site data. To support the future request for license termination, the site survey results, soil maps, and methods used to revise Kd are carefully documented.

The revised source term and parameter values are used in iteration 2 of the dose assessment in step 4. In this example, the licensee decides to leave the original default model assumptions and pathways unchanged, and continues to use the DandD software. [Note that in other more complicated situations a licensee might seek to modify these assumptions and pathways. A detailed submittal discussing such changes would need to be developed]. When the revised parameter values are input into the model, the result is a dose equal to 25 mrem/y.

This brings the licensee back to step 5 and the question regarding whether the site can be released. Since the dose assessment result is equal to 25 mrem/y, and the site survey met the minimum requirements for a final release survey, the licensee can move on to consider any remaining ALARA requirements. The licensee can document that best practice procedures were applied as part of its operational program. In addition, ALARA was incorporated and documented in the options definition (step 8), analysis of options (step 9), and selection of the preferred option (step 10).

Based on the above, the license can be terminated and the site released. The licensee submits all required forms, including NRC Form 314, and documentation of the decision process, and the site is released for unrestricted use.

An Overview of NRC Risk-Informed, Performance-Based Initiatives

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The NRC's research program in probabilistic risk analysis consists of a set of closely-related elements, from basic research to regulatory applications. The objectives of this research program are linked to specific agency regulatory functions, and are as follows:

- To develop, demonstrate and improve methods for assessing the risks of <u>nuclear power plant operations</u>
- To develop and demonstrate risk assessment methods applicable to <u>non-reactor facilities and operations</u> licensed for the production, processing, or utilization of radioactive materials in industrial, medical, and academic applications
- To review licensees' IPE and IPEEE submittals to determine if they meet the intent of NRC's Generic Letter 88-20

This paper provides an overview of major programs supporting each of these objectives.

I. Introduction

The NRC's research program in probabilistic risk analysis consists of a set of closely-related elements, from basic research to regulatory applications. The objectives of this research program are linked to specific agency regulatory functions, and are as follows:

- To develop, demonstrate and improve methods for assessing the risks of <u>nuclear power plant operations</u> which will facilitate their use in implementing the Commission's goal of improved regulatory effectiveness and risk-informed regulatory decision-making; and to develop and promulgate appropriate guidance documents for use by both licensees and staff in uniformly applying risk assessment methods to support agency-wide decision-making, to support evaluation a licensee's requests for license amendments, and to support staff assessments of the significance of abnormal operating events.

- To develop and demonstrate risk assessment methods applicable to <u>non-reactor facilities and operations</u> licensed for the production, processing, or utilization of radioactive materials in industrial, medical, and academic applications, to develop appropriate guidance documents for the use of such methods in risk-informed regulatory applications, and to provide as-needed support for NMSS in the application of risk assessment technology in its regulatory functions.

- To review licensees' IPE and IPEEE submittals to determine if they meet the intent of NRC's Generic Letter 88-20 (IPE) (Ref. 1) and Supplement 4 to Generic Letter 88-20 (IPEEE) (Ref. 2), and to analyze information from the review of licensees' IPE and IPEEE submittals to provide generic perspectives and insights from these programs.

Major programs supporting each of these objectives are summarized below. Other papers in the session provide additional detail on specific, key programs.

II. Develop methods and guidance for nuclear power plants

Methods

Human Reliability Analysis

It has been accepted for some time that failures in human performance are one of the principal sources of risk. Although techniques have been used in the past to quantify both pre-accident and post-accident human error, one of the remaining questions is how to treat "errors of commission." The NRC and its contractors are developing methods for treating human errors of commission (Refs. 3 and 4). The general process will be to:

- Identify potentially unsafe actions and reasons for human failure events,
- Identify potential significant error forcing contexts (those conditions that "conspire" to cause operators to take unsafe actions), and
- Estimate the likelihood of potentially significant error forcing contexts and unsafe actions.

Other papers in this conference describe this work in more detail.

Fire Risk Analysis

Since being prompted by the Browns Ferry fire of 1975, a number of nuclear power plant fire risk assessments have shown that fires can be significant contributors to plant risk. The most important scenarios identified in these analyses tend to involve the occurrence of relatively infrequent fires whose location and severity are such that critical sets of plant equipment are likely to be damaged by such a fire, if it occurs. These general conclusions regarding the potential magnitude and character of nuclear power plant fire risk appear to be consistent with empirical evidence, where serious fire-induced challenges to reactor core cooling are not common events but have occurred.

While there is little argument about the potential importance of fires, the magnitude of the fire risk and the specific measures needed to efficiently manage this risk are not as clear when considering individual plants. The variability in the estimated fire risk and risk contributors is due not only to plant-specific variations in design and operation, but also to variations in the methods and data used in the studies. Uncertainties in the current state of knowledge concerning the initiation, growth, suppression, and plant impacts of fire-induced nuclear power plant accident scenarios all contribute to this latter category of variability; they have raised significant concerns regarding the usefulness of current fire risk assessment tools in support of proposed plant changes and the development of a risk-informed, performance-based rule for nuclear power plant fire protection.

In response to these concerns, NRC is initiating a research program to develop and demonstrate improved methods for performing fire risk assessment. This program is the subject of a more detailed paper in this session.

Accident Sequence Precursors

The NRC routinely evaluates operational events for safety significance and generic implications. Since the late 1970s, probabilistic analysis techniques have been used in such evaluations. This provides quantitative evaluations and also enforces a disciplined, consistent approach to event analysis.

As PRA techniques have evolved and increased in sophistication, so have these evaluations. During the past several years, 75 new "simplified plant analysis risk" (SPAR) models have been developed to represent virtually all plants in the country (Ref. 5). These models are designed to run with the most recent (Windows NT) version of NRC's SAPHIRE computer software (Ref. 6).

The NRC plans to improve the SPAR models in a number of respects over the next several years:

- The 75 models will be modified to include plant-specific dependencies and other features based on a review of the Individual Plant Evaluations and on responses to the Station Blackout Rule.
- External event analyses (seismic, fire, flood) will be added to the SPAR models.
- The models will be extended to consider low power/shutdown conditions.
- The models will be expanded to reflect public health consequences and risk, using, for example, a surrogate metric - large early release frequency. This will be done to more correctly evaluate the significance of events which could involve relatively high consequences (e.g., containment bypass scenarios).

<u>Guidance</u>

The NRC has issued for public comment drafts of a set of regulatory guides (RGs) and standard review plan (SRP) sections. The guidance is for power reactor licensees, and provides acceptable methods for using probabilistic risk assessment (PRA) information in support of applications to change a licensed plant's current licensing basis (CLB). Currently, draft RGs/SRPs (Refs. 7 through 15) have been developed and issued for comment in the areas of general guidance, inservice testing (IST), inservice inspection (ISI), technical specifications (TS), and graded quality assurance (GQA).

The RGs describe one acceptable means by which licensees can propose plant-specific CLB changes under 10CFR50. Licensees submitting applications for changes to their CLB may use this approach or an alternative equivalent approach. A general RG (DG-1061) and SRP (Refs. 7 and 8) have been developed to provide an overall framework and guidance that is applicable to any proposed CLB change where risk information is used to support the change. The application-specific RGs/SRPs (i.e., IST, ISI, TS, GQA) build upon and supplement the general guidance for proposed CLB changes in their respective technical areas.

In conjunction with developing these RGs and SRPs, the staff has also been working with several licensees on pilot applications of risk informed regulation in IST, ISI, TS, and GQA. The knowledge gained to date in interacting with licensees on these pilot applications has been used to help refine the content of and guidance contained in these RGs/SRPs.

The fundamental principals for the staff's approach for using PRA in risk-informed CLB changes are described in the draft general regulatory guide. A separate paper in this session provides additional information on this regulatory guide.

III. Develop and demonstrate methods for non-reactor facilities

As the NRC moves toward a risk-informed approach to regulation, probabilistic risk analysis tools are being put to use across the entire spectrum of the Agency's work. Although most PRAs are studies of nuclear reactor safety, the same techniques can potentially be applied to nuclear materials safety and associated issues. Two programs to develop and apply such techniques currently exist within the Office of Nuclear Regulatory Research, related to spent fuel, dry cask storage facilities and sealed radioactive sources used in industrial facilities. These programs are the subject of a separate paper in this session.

III. Review IPEs and IPEEEs

In 1985, NRC issued its "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants" (Ref. 16) that introduced the Commission's plan to address severe accident issues for existing commercial nuclear power plants. In this policy statement, the Commission addressed its plan to formulate an approach for a systematic safety examination of existing plants to study particular accident vulnerabilities and desirable cost-effective changes so as to ensure that there is no undue risk to public health and safety. NRC's Generic Letter (GL) 88-20 (Ref. 1) requested all licensees to perform an individual plant examination (IPE) to identify any plant-specific vulnerabilities to severe accidents, and to report the results to the Commission. Supplement 4 requested licensees to perform an IPE of external events and also report these results to the Commission (Ref. 2).

As a result of GL 88-20, 75 IPE submittals were received from the licensees covering 108 units and 74 IPEEE submittals are being received from the licensees covering 107 units (some licensees elected not to perform an IPEEE). Key activities in the staff's IPE and IPEEE review process include:

- <u>Review program</u> Each IPE and IPEEE submittal is reviewed with a focus on whether the licensee's method was capable of identifying vulnerabilities, and therefore meets the intent of GL 88-20. The review considers (1) the completeness of the information and (2) the reasonableness of the results given the plant design, operation, and history. The staff has now completed essentially all IPE reviews, with staff evaluation reports issued to licensees. With respect to the review of the IPEEE submittals, the staff has received more than two-thirds of the submittals; the staff's reviews of these submittals is scheduled to be completed in mid-1999.

- <u>Insights program</u> This program collects and documents the significant safety insights, based on the IPEs and IPEEEs, for the different reactor and containment types and plant designs. With respect to the IPEs, there are five major objectives which involve providing perspectives on the following:

    - Impact of the IPE Program on Reactor Safety: perspectives on the number and type of vulnerabilities or safety issues, impact of the safety enhancements, and the generic applicability of the vulnerabilities and safety enhancements.
    - Reactor and Containment Design Perspectives: perspectives on the important design and

operational features, methods and assumptions, and significant plant improvements that affect the core damage frequency and containment performance for different reactor and containment types.

- Importance of the Operator's Role: perspectives on operator actions either consistently found important across the IPEs or found important due to plant-specific characteristics, on the influence of modeling assumptions and different methodologies, and on the causes of the variability in CDF estimation and containment performance analysis.
- IPEs with Respect to Risk-Informed Regulation: perspectives on the IPEs in risk-informed regulation.
- Perspectives On Some Additional Items: (a) perspectives on the IPE results relative to the Commission's Safety Goals; (b) perspectives on the improvements that have been identified as a result of the Station Blackout Rule and analyzed as part of the IPE and the impact of these improvements on reducing the likelihood of station blackout; and (c) perspectives of the IPEs as compared to the perspectives gained from NUREG-1150 (Ref. 17).

The results of the IPE insights program were issued in late 1997 as Reference 18. Preliminary insights from the reviews of the first IPEEEs were also issued in early 1998 as Reference 19.

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Development of Risk-Informed Regulatory Guidance to Industry and the NRC Staff

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Abstract

The Nuclear Regulatory Commission has issued for public comment drafts of regulatory guides, standard review plan sections, and a NUREG document. These issuances follow publication of the Commission's August 16, 1995, "Policy Statement on the Use of PRA Methods in Nuclear Regulatory Activities." The regulatory guides and standard review plan sections are intended to provide an acceptable approach for power reactor licensees to prepare and submit, and NRC staff to review, applications for proposed plant-specific changes to the plant's current licensing basis that utilize risk information. Draft documents have been developed and issued for public comment in the areas of general guidance, inservice testing, inservice inspection, technical specifications, and graded quality assurance. This paper summarizes the scope of this effort and then focuses on some of the key general characteristics of the approach.

I. Introduction

The Nuclear Regulatory Commission has issued for public comment drafts of a set of regulatory guides (RGs) and standard review plan (SRP) sections, and a supporting staff report. These issuances follow publication of the Commission's August 16, 1995, "Policy Statement on the Use of PRA Methods in Nuclear Regulatory Activities" (Ref. 1). The guidance is for power reactor licensees, and provides acceptable methods for using probabilistic risk assessment (PRA) information in support of applications to change a licensed plant's current licensing basis (CLB). Currently, draft RGs/SRPs (Refs. 2 through 10) have been developed and issued for comment in the areas of general guidance, inservice testing (IST), inservice inspection (ISI), technical specifications (TS), and graded quality assurance (GQA).[1] In addition, the NRC has prepared and issued for public comment a draft report on the attributes of a PRA used in risk-informed applications to provide reference information for licensees and NRC staff (Ref. 11).

The RGs describe one acceptable means by which licensees can propose plant-specific CLB changes under 10CFR50. Licensees submitting applications for changes to their CLB may use this approach or an alternative equivalent approach. To encourage the use of risk information in such applications, the staff intends to give priority to applications for burden reduction that use risk information as a supplement to traditional engineering analyses, consistent with the intent of the Commission's policy. All applications that improve safety will continue to receive high priority.

[1] No SRP has been developed for GQA, since the NRC staff will utilize its inspection process in this area.

A general RG and SRP (Refs. 2 and 3) have been developed to provide an overall framework and guidance that is applicable to any proposed CLB change where risk information is used to support the change. The application-specific RGs/SRPs (i.e., IST, ISI, TS, GQA) build upon and supplement the general guidance for proposed CLB changes in their respective technical areas. Each application-specific RG/SRP references the general RG/SRP, states that the general guidance is applicable, and provides additional guidance specific to the technical area being addressed.

In conjunction with developing these RGs and SRPs, the staff has also been working with several licensees on pilot applications of risk informed regulation in IST, ISI, TS, and GQA. The knowledge gained to date in interacting with licensees on the pilot applications has been used to help define the content and guidance contained in these RGs/SRPs. Additional interactions are expected over the next several months as work on these pilot applications continues and licensees and other interested persons have an opportunity to review the draft RGs/SRPs during the public comment period. The results of these additional interactions will be factored into the final RGs/SRPs, which are scheduled to issued in early 1998.

The fundamental principals for the staff's approach for using PRA in risk-informed CLB changes are described in the draft general regulatory guide (DG-1061) (Ref. 2). As such, the remainder of this paper provides additional information on this regulatory guide.[2]

II. General Guidance for Using PRA in Risk-Informed CLB Changes

<u>Principles and Expectations</u>

In implementing risk-informed decision-making, proposed CLB changes are expected to meet a set of key principles. Some of these principles are written in terms typically used in traditional engineering decisions (e.g., defense-in-depth). While written in these terms, risk analysis techniques can be, and are encouraged to be, used to help ensure and show that the principles are met. These principles are:

1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change.

2. The proposed change is consistent with the defense-in-depth philosophy.

3. The proposed change maintains sufficient safety margins.

4. When proposed changes result in an increase in core damage frequency and/or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

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[2] At the time of the writing of this paper, the staff's general regulatory guide had evolved beyond draft DG-1061. Section II of this paper reflects the staff's guidance contained in the proposed final version of the guide, Regulatory Guide 1.174, as provided to the Commission for approval for publication in Reference 12.

5. The impact of the proposed change should be monitored using performance measurement strategies.

The DG-1061 evaluation approach and acceptance guidelines follow from these principles. In implementing these principles, it is expected that:

- All safety impacts of the proposed change are evaluated in an integrated manner as part of an overall risk management approach in which the licensee is using risk analysis to improve operational and engineering decisions broadly by identifying and taking advantage of opportunities for reducing risk, and not just to eliminate requirements the licensee sees as undesirable. For those cases where risk increases are proposed, the benefits should be described and should be commensurate with the proposed risk increases. The approach used to identify changes in requirements should be used to identify areas where requirements should be increased, as well as where they could be reduced.

- The scope and quality of the engineering analyses (including traditional and probabilistic analyses) conducted to justify the proposed CLB change should be appropriate for the nature and scope of the change, should be based on the as-built and as-operated and maintained plant, and should reflect operating experience at the plant.

- The plant-specific PRA supporting licensee proposals have been subjected to quality controls such as an independent peer review or certification.

- Appropriate consideration of uncertainty is given in analyses and interpretation of findings, including using a program of monitoring, feedback, and corrective action to address significant uncertainties.

- The use of core damage frequency (CDF) and large early release frequency (LERF)[3] as bases for probabilistic risk assessment acceptance guidelines is an acceptable approach to addressing Principle 4. Use of the Commission's Safety Goal quantitative health objectives (QHOs)(Ref. 13) in lieu of LERF is acceptable in principle and licensees may propose their use. However, in practice, implementing such an approach would require an extension to a Level 3 PRA, in which case the methods and assumptions used in the Level 3 analysis, and associated uncertainties, would require additional attention.

- Increases in estimated CDF and LERF resulting from proposed CLB changes will be limited to small increments; the cumulative effect of such changes should be tracked and considered in the

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[3] In this context, LERF is being used as a surrogate for the safety goal early fatality QHO. It is defined as the frequency of those accidents leading to significant, unmitigated releases from containment in a time frame prior to effective evacuation of the close-in population such that there is a potential for early health effects. Such accidents generally include unscrubbed releases associated with early containment failure at or shortly after vessel breach, containment bypass events, and loss of containment isolation. This definition is consistent with accident analysis used in the safety goal screening criteria discussed in the Commission's Regulatory Analysis Guidelines (Ref. 14).

decision process.

- The acceptability of proposed changes should be evaluated by the licensee in an integrated fashion that ensures that all principles are met.

- Data, methods, and assessment criteria used to support regulatory decision-making must be well documented and available for public review.

Evaluation Process

A four-element approach to evaluating proposed CLB changes, consistent with the principles of risk-informed decision-making discussed above, is used in DG-1061 to define an acceptable approach for using PRA in decision making. The four elements are as follows:

Element 1: Define the Proposed Change

Element 1 requires a specification of three types of information: first, the identification of those aspects of the plant's licensing bases that may be affected by the proposed change, including, but not limited to, rules and regulations, final safety analysis report, technical specifications, licensing conditions, and licensing commitments; second, identification of all systems, structures, and components (SSCs), procedures, and activities that are covered by the CLB change under evaluation and consider the original reasons for inclusion of each program requirement; third, identification of available engineering studies, methods, codes, applicable plant-specific and industry data and operational experience, PRA findings, and research and analysis results relevant to the proposed CLB change. With particular regard to the plant-specific PRA, the capability to use, refine, augment, and update system models as needed to support a risk assessment of the proposed CLB change should be assessed. The above information should be used collectively to provide a description of the CLB change and to outline the method of analysis.

Element 2: Perform Engineering Analysis

The proposed CLB change should be evaluated with regard to the principles that adequate defense-in-depth is maintained, that sufficient safety margins are maintained, and that proposed increases in core damage frequency and risk are small and are consistent with the intent of the Commission's Safety Goal Policy Statement (Ref. 12). The scope and quality of the engineering analyses conducted to justify the proposed CLB change should be appropriate for the nature and scope of the change, and appropriate consideration should be given to uncertainty in the analysis and interpretation of findings.

Element 3: Define Implementation and Monitoring Program

Careful consideration should be given to implementation and performance-monitoring strategies. The primary goal for this element is to ensure that no adverse safety degradation occurs because of the changes to the CLB. The principal concern is the possibility that the aggregate impact of changes which affect a large class of SSCs could lead to an unacceptable increase in the number of failures due to unanticipated degradation, including possible increases in common cause mechanisms. Therefore, an implementation and monitoring plan should be developed to ensure that the assumptions underlying the engineering evaluation conducted to examine the impact of the proposed changes are justified. This will ensure that the conclusions which have been drawn from the evaluation remain valid.

Decisions concerning implementation of changes should be made in light of the uncertainty associated with the results of the traditional and probabilistic engineering evaluations. Broad implementation within a limited time period may be justified when uncertainty is shown to be low (data and models are adequate, engineering evaluations are verified and validated, etc.), whereas a slower, phased approach to implementation (or other modes of partial implementation) would be expected when uncertainty in evaluation findings is higher and where programmatic changes are being made which potentially impact SSCs across a wide spectrum of the plant, such as in IST, ISI and graded QA. In such situations, the potential introduction of common cause effects must be fully considered and included in the submittal.

The staff expects licensees to propose monitoring programs that include a means to adequately track the performance of equipment which, when degraded, can affect the conclusions of the licensee's engineering evaluation and integrated decision-making that support the change to the CLB. The program should be capable of trending equipment performance after a change has been implemented to demonstrate that performance is consistent with that assumed in the traditional engineering and probabilistic analyses that were conducted to justify the change. This may include monitoring associated with non-safety related SSCs, if the analysis determines those SSCs to be risk significant. The program should be structured such that: (1) SSCs are monitored commensurate with their safety importance, i.e., monitoring for SSCs categorized as low safety significant may be less rigorous than that for SSCs of high safety significance; (2) feedback of information and corrective actions are accomplished in a timely manner; (3) degradation in SSC performance is detected and corrected before plant safety can be compromised. The potential impact of observed SSC degradation on similar components in different systems throughout the plant should be considered.

Element 4: Submit Proposed Change

DG-1061 discusses the documentation required when submitting the request for a change for review and approval by NRC. With respect to the PRA information to be submitted, it is necessary to describe how, and to what extent, the impact of the change has been incorporated in the PRA model. It is not only the numerical results of the PRA that is required, but also an analysis of the contributors to those results that are necessary for the decision, and a discussion of why the decision is appropriate in light of the analytical uncertainties.

DG-1061 provides acceptance guidelines for both the engineering and risk evaluations performed in Element 2, described above. The remainder of this section focuses on the risk acceptance guidelines These guidelines have been established to be consistent with the principles and expectations for risk-informed regulation discussed above. The guidelines are intended for comparison with a full scope (including internal events, external events, full power, low power and shutdown) assessment of the change in risk metric, and, when necessary, as discussed below, the baseline value of the risk metric (CDF or LERF). However, it is recognized that many PRAs are not full scope and the use of less than full scope quantitative PRA information may be acceptable.

<u>Probabilistic Acceptance Guidelines</u>

There are two probabilistic acceptance guidelines, one for CDF and one for LERF, both of which should be used.

The guidelines for CDF are:

- If the application can be clearly shown to result in a decrease in CDF, the change will be considered to have satisfied the relevant principle of risk-informed regulation with respect to CDF.

- When the calculated increase in CDF is very small, which is taken as being less than 10^{-6} per reactor year, the change will be considered regardless of whether there is a calculation of the total CDF. While there is no requirement to calculate the total CDF, should there be an indication that the CDF may be considerably higher than 10^{-4} per reactor year, the focus should be on finding ways to decrease rather than increase it. Such an indication would result, for example, if: (1) the contribution to CDF calculated from a limited scope analysis, such as the IPE, and, if appropriate the IPEEE, significantly exceeds 10^{-4} per reactor year; (2) there has been an identification of a potential vulnerability from a margins type analysis; or (3) historical experience at the plant in question has indicated a potential safety concern.

- When the calculated increase in CDF is in the range of 10^{-6} per reactor year to 10^{-5} per reactor year, applications will be considered only if it can be reasonably shown that the total CDF is less than 10^{-4} per reactor year.

- Applications which result in increases to CDF above 10^{-5} per reactor year would not normally be considered.

AND

The guidelines for LERF are:

- If the application can be clearly shown to result in a decrease in LERF, the change will be considered to have satisfied the relevant principle of risk-informed regulation with respect to LERF.

- When the calculated increase in LERF is very small, which is taken as being less than 10^{-7} per reactor year, the change will be considered regardless of whether there is a calculation of the total LERF. While there is no requirement to calculate the total LERF, should there be an indication that the LERF may be considerably higher than 10^{-5} per reactor year, the focus should be on finding ways to decrease rather than increase it. Such an indication would result, for example, if: (1) the contribution to LERF calculated from a limited scope analysis, such as that the IPE, and, if appropriate the IPEEE, significantly exceeds 10^{-5} per reactor year; (2) there has been an identification of a potential vulnerability from a margins type analysis; or (3) historical experience at the plant in question has indicated a potential safety concern.

- When the calculated increase in LERF is in the range of 10^{-7} per reactor year to 10^{-6} per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 10^{-5} per reactor year.

- Applications which result in increases to LERF above 10^{-6} per reactor year would not normally be considered.

These guidelines are intended to provide assurance that proposed increases in CDF and LERF are small and are consistent with the intent of the Commission's Safety Goal Policy Statement. To ensure this consistency, DG-1061 includes a discussion of how PRA results are to be calculated, including discussion on use of mean CDF and LERF values and the consideration of uncertainties. A key element of this discussion is that the total plant CDF and LERF are to be used for comparison with the acceptance guidelines described above, although such comparisons can be quantitative or qualitative.

In addition, DG-1061 includes a general description of the attributes of the PRA necessary for use in risk-informed CLB changes. Some of the key elements of this description include:

- The scope, level of detail, and quality required of the PRA should be commensurate with the application for which it is intended and on the role the PRA results play in the integrated decision process. The more emphasis that is put on the risk insights and on PRA results in the decision-making process, the more requirements have to be placed on the PRA, both in terms of scope and in terms of how well the risk and/or the change in risk is assessed.

- The PRA performed should realistically reflect the actual design, construction, operational practices, and operational experience of the plant and its owner. This should include licensee voluntary actions as well as regulatory requirements and the PRA used to support risk-informed decision making should also reflect the impact of previous changes made to the CLB.

- The level of detail required of the PRA is that which is sufficient to model the impact of the proposed change. The characterization of the problem should include the establishment of a cause-effect relationship to identify portions of the PRA affected by the issue being evaluated.

III. Conclusions

When finalized, the staff's risk-informed regulatory guides and standard review plan sections will define a consistent set of principles and practices by which PRA information can be used in the analysis of proposed changes to plant's current licensing basis. This guidance integrates long-standing traditional engineering analyses with risk assessment information and Commission safety goal policies. These guides provide a key element of the NRC's activities to make its regulatory processes more risk-informed, as directed by the Commission's PRA Policy Statement.

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AN UPDATE OF PRELIMINARY PERSPECTIVES GAINED FROM INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) SUBMITTAL REVIEWS

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ABSTRACT

As a result of the U.S. Nuclear Regulatory Commission (USNRC) initiated Individual Plant Examination of External Events (IPEEE) program, virtually every operating commercial nuclear power reactor in the United States has performed an assessment of severe accident risk due to external events. To date, the USNRC staff has received 63 IPEEE submittals and will receive an additional 11 by mid 1998. Currently, 49 IPEEE submittals are under various stages of review. This paper is based on the information available for those 41 plants for which at least preliminary Technical Evaluation Reports have been prepared by the review teams. The goal of the review is to ascertain whether the licensee's IPEEE process is capable of identifying external events-induced severe accident vulnerabilities and cost-effective safety improvements to either eliminate or reduce the impact of these vulnerabilities. The review does not, however, attempt to validate or verify the results of the licensee's IPEEE. The primary objective of this paper is to provide an update on the preliminary perspectives and insights gained from the IPEEE process.

INTRODUCTION

In 1988 the USNRC requested through GL 88-20 [1] that all licensees conduct an individual plant examination (IPE) of severe accident risk for internally initiated events and report the results to the USNRC. In a follow-on to this initial request, on June 28, 1991, the USNRC issued Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, 10 CFR 50.54(f)," [2] requesting all licensees to perform a complementary assessment to identify plant-specific vulnerabilities to severe accidents caused by external events. Attached to Supplement 4 was NUREG-1407 "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities: Final Report" [3]. This report provided licensees with more information on the scope of the IPEEE and outlined the information that should be submitted by licensees. Finally, in order to address issues related to the 1993 Lawrence Livermore National Laboratory (LLNL) seismic hazard curves [4], on September 8, 1995, Supplement 5 to GL 88-20 [5] was issued to provide guidance on modifying the scope of the seismic IPEEE for certain plants.

The purpose of this paper is to provide an update on the current status of the IPEEE process, and to document preliminary perspectives, findings, and lessons learned deriving from the results of the IPEEE reviews that have been undertaken to date. The major areas of the IPEEE analyses include the following external event initiators: seismic events, fires, and high winds, floods, and other initiators (HFO).

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IPEEE OBJECTIVES AND THE REVIEW PROCESS

The overall goal of the IPEEE as established in GL 88-20 Supplement 4 is for licensees to identify external events-induced severe accident vulnerabilities and cost-effective safety improvements to either eliminate or reduce the impact of these vulnerabilities. To reach this goal, four IPEEE objectives were established as follows:

1. to develop an appreciation of severe accident behavior,

2. to understand the most likely severe accident sequences that could occur at its plant under full power operating conditions,

3. to gain a qualitative understanding of the overall likelihood of core damage and fission product releases, and

4. if necessary, to reduce the overall likelihood of core damage and radioactive material releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

A total of 63 plant IPEEE submittals have been received to date by the USNRC. Of these, 49 have either been reviewed, or are currently undergoing review. The objective of the review process has been to determine whether each licensee submittal has met the intent of the IPEEE process as characterized by the four objectives listed immediately above. However, it is not the intent of the reviews to validate or verify the results of the licensee's IPEEE.

As originally conceived, the review process was comprised of a "Step 1" review of each submittal, and follow-on "Step 2" reviews of individual submittals on an "as needed" basis. The Step 1 review is based on a review of the submittal only. This means that none of the underlying or supporting (second tier) documents are examined. This review step also includes interactions with the licensee through the Request for Additional Information (RAI) process, conference calls, or public meetings. The objective of the licensee interactions is to obtain clarification of specific points in the submittal which were either unclear or of questionable basis. These RAIs have generally been limited to items considered to be of sufficient importance that the insights or findings of the IPEEE, or the reviewers understanding of those findings and insights, might be significantly impacted by the licensee response.

If, at the end of the Step 1 review process, it cannot be concluded that a given submittal has met the intent of the IPEEE process or unusual results (i.e., extremely high or low CDFs/ HCLPFs) were reported, then a Step 2 review may be undertaken. A Step 2 review would typically include further licensee interactions (i.e., review of supporting second tier documents, a plant visit, interviews with the plant personnel, and plant walkdown) to resolve those concerns. Thirty of the 49 plant reviews initiated to date have been performed as "Step 1" reviews. Of the 30 submittals for which Step 1 reviews have been performed, one has already undergone a Step-2 review and six more have been recommended for some level of further review (Step 2).

Because of the USNRC's budget constraint and the experience gained in the IPE submittal reviews, in late 1996 an alternate process of screening reviews was implemented. The objective of the screening reviews is to identify those licensee submittals that have clearly met the intent of the IPEEE process. The screening reviews are somewhat more limited in scope than a corresponding Step 1 review, and less effort is expended in reviewing the submittals and in documenting the review findings. Experience to date indicates that the majority of these screening reviews will require some limited scope RAIs before a final judgement on meeting the intent of the process can be made. The scope of these RAIs includes key aspects of the analysis that have not been adequately documented and/or apparent mistakes or oversights with the potential to fundamentally impact the licensee's results. Currently, there are 19 submittals for which screening reviews have been initiated (this paper includes findings for the first 11 of these 19 screening reviews).

As a part of the review process, the USNRC has also convened a Senior Review Board (SRB) to oversee the technical aspects of the review process. The SRB is comprised of USNRC staff and contractors who are expert in the fields of general risk assessment and the specific areas covered by the IPEEE analyses (seismic, fire, and HFOs). The SRB members also perform abbreviated reviews of each of the IPEEE submittals. Regular meetings are then held at which the contractor reviewers with primary responsibility for the review of a given plant submittal present their own findings, insights, and recommendations. The SRB then comments on the completeness of the review, whether the reviewer's technical findings are of sufficient importance to warrant a RAI to the licensee, and whether the submittal has met the IPEEE intent. The SRB participates in all levels of review - Screening, Step 1, and Step 2.

The review process has confirmed that most of the IPEEEs have involved substantial effort on the part of licensees. It has also revealed considerable variability in the selection of methods for analysis, the underlying assumptions and inputs used, and the approach to and level of risk quantification employed (e.g., screening versus detailed quantification). While virtually all of the submittals have followed the general reporting outline recommended in NUREG-1407, within this framework there has been considerable variation in the approach to reporting, and the level of detail provided in the submittals. These factors have somewhat complicated the review process because each submittal is substantially different.

SEISMIC PERSPECTIVES

On the basis of relative ranking of seismic hazards, the staff has designated nuclear power plant sites into the following seismic evaluation categories (ref: Supplement 4 to Generic Letter 88-20 and NUREG-1407):

Eastern United States (East of the Rocky Mountains) Plant Sites

1. Reduced-scope
2. 0.3g Focused-scope
3. 0.3g Full-scope
4. Seismic PRA (Licensees committed to perform a seismic PRA)

Western United States Plant Sites

5. Seismic margin methods (0.3g Full-scope and 0.5g)
6. Seismic PRA

As described in NUREG-1407, a seismic PRA methodology is acceptable for plants in all evaluation categories; however, a seismic margin assessment (SMA) is also acceptable for plants in evaluation categories 1, 2, 3, and 5. Of the 41 seismic IPEEEs reviewed to date, 18 used a seismic PRA (SPRA), while the remaining 23 used a seismic margin assessment. Among those SPRAs, a limited number of submittals have used a hybrid approach in which the initial SMA screening procedures were used in conjunction with the risk quantification.

Plant Seismic Core Damage Frequency (CDF)

Table 1 summarizes the seismic CDF results obtained from the 18 SPRA submittals reviewed to date. Seismic CDFs are observed to range from less than 1E-07 per reactor year (ry) to 2.3E-04/ry. This broad variation cannot be attributed to the use of different seismic hazard curves (e.g., Lawrence Livermore National Laboratory (LLNL) curves versus Electric Power Research Institute (EPRI) curves), since some higher seismic CDF values are based on the EPRI seismic hazard curves. Rather, the broad variation is due to differences in seismic hazard levels at the different site locations in conjunction with differing levels of as-designed seismic capacities. In addition, differences in assumptions and modeling used for the SPRA quantification also contribute significantly to the broad range of observed CDF variation.

Plant Seismic Capacity

For the three Western U.S. plants reviewed to date, two had high confidence, low probability of failure (HCLPF) values (in terms of peak ground acceleration, PGA) of 0.67g, while the third estimated a HCLPF of 0.50g. For plant sites east of the Rocky Mountains, the plant HCLPF values derived from seismic PRAs have ranged from less than 0.05g to 0.50g.

In addition to a value of PGA (or other parameter), a spectral shape is also needed to define a plant HCLPF capacity. As Table 1 indicates, seismic capacity results developed from seismic PRAs are most often associated with a site-specific spectral shape, usually the uniform hazard spectra developed in the LLNL or EPRI hazard programs. In some cases, the NUREG/CR-0098 spectral shape has been used for evaluating seismic capacity.

Table 2 presents a list of plant-level HCLPF results reported in the licensees' SMA IPEEEs reviewed to date. All plants in this list are located east of the Rocky Mountains with the exception of Palo Verde in Arizona. As seen, 3 full-scope, 14 focused-scope and 7 reduced-scope submittals have been reviewed. (As noted, three sites assigned to the focused-scope category elected to perform assessments at the reduced-scope level).

The plant HCLPF capacities for the full-scope and focused-scope plants are seen to vary from 0.09g to 0.50g. For the full-scope and focused- scope plants, HCLPF values presented in Table 2 have been derived based on a NUREG/CR-0098 median spectral shape for rock or soil (depending on the site conditions at the plant). (Reduced-scope plants verified their seismic adequacy at their design basis earthquake level, SSE or DBE and associated design spectra, and were not expected to compute a plant HCLPF).

Walkdown Insights

The vast majority of the SPRA and SMA IPEEEs reviewed to date have stated that EPRI NP-6041 [6] procedures were used for performing seismic screening and walkdowns. For Unsolved Safety Issue (USI) A-46 plants, the walkdown procedures and criteria described in the generic implementation procedure [7] were used in all seismic IPEEEs.

In general, anomalous conditions for plants revealed from the walkdown efforts were related to the following items:

- adequacy of equipment anchorage
- quality of installation
- physical interactions
- seismic maintenance and housekeeping

Relay Evaluation

NUREG-1407 describes the recommended procedures for relay evaluation, depending on the scope of seismic evaluation and on whether or not the plant is a USI A-46 plant. Relay evaluations for USI A-46 plants have revealed low ruggedness (denoted as "bad actor") relays at a number of plants. However, beyond the selected USI A-46 safe shutdown paths, only a few of these plants assessed to date have encountered bad actor relays in other safe shutdown paths selected for IPEEE. For non-USI A-46 plants assessed to date, relay evaluations have revealed a few bad actor relays at a majority of the plants.

When bad actor relays have been encountered, they have often been found to exist in alarm circuitry, they have been assessed as having negligible consequences resulted from the effect of relay chatter, or they have been determined that operator actions will be able to reset the function of these relays. Consequently, in only a few isolated instances have licensees proposed to replace these bad actor relays.

Soils Evaluation

Most licensees whose plants are identified as soil sites (and are not in the reduced-scope seismic category) have provided information addressing the issue of soil failure effects in their IPEEE submittals. A few licensees, who made use of the modified seismic IPEEE guidelines described in Supplement 5 to GL 88-20, have not provided a soils evaluation in their IPEEEs.

In one case, liquefaction was predicted at a level not too much greater than the SSE, and in this case, liquefaction was found to be the dominant failure mode for several important structures and fire water piping. In two other cases, the soils evaluation has indicated that liquefaction is likely to occur at the review level earthquake (RLE) and seismic slope instability is likely to occur at the RLE for these two plants. However, the magnitude of slope deformations was assessed as being minor. For most soil site submittals, however, the impacts of seismic-induced soil settlements and soil deformations have been assessed as being minor.

Non-Seismic Failures and Human Actions

All IPEEEs assessed to date have provided some discussion of non-seismic failures and human actions. For SPRA IPEEEs, these effects have been introduced in seismic event-tree and fault-tree models which have been based on plant logic constructed for internal events. It is important to note, though, that seismic impacts on operator error rates have been modeled in a wide variety of fashions among the IPEEE submittals assessed to date. In some seismic PRAs, simplified operator error fragilities have been developed. In other instances, debatable scaling factors on internal event error rates have been applied based on the importance of the human action or on other factors. A notable insight is that, when operator error fragilities have been applied, they often acted to mask the seismic failures that dominate seismic CDF. Operator fragilities are highly uncertain; hence, it is important to identify the specific operator actions and undertake a sensitivity study to reveal the relative significance of seismic failures and their impact on operator actions.

In only a few cases have screening criteria been actually applied with respect to random failure rates and human error rates. Most frequently, the SMA IPEEE submittals assessed to date have simply reported an attempt to rely on those seismic success paths that are most familiar to plant operators and that utilize the most reliable equipment.

Seismic-Fire Evaluation

All IPEEEs assessed to date have attempted to evaluate the following seismic-fire interaction issues:

- seismic-initiated fires
- seismic actuation of fire suppression systems

However, the treatment on these issues are rather diversified; some submittals have evaluated them thoroughly in certain areas while other submittals are less thorough. Perhaps most consistently, however, the following are noticed:

- the locations of fire sources have often not been clearly identified
- seismic-induced flooding due to sources other than fire water piping (e.g., tank failures and non-fire-water piping) has often been neglected

The most consistent strong points of the seismic-fire evaluations appear to be the treatment of inadvertent actuation of fire suppression systems and the identification of potential interaction concerns. A number of the IPEEE submittals have produced some significant findings and have resulted in some plant-specific improvements as discussed further below.

Dominant Risk Contributors

In most instances, dominant risk contributors (seismic failures, random failures, and operator errors), that may lead to core damage, are identified in these SPRA IPEEEs. The following dominant contributors have been reported to be of most significance to seismic CDF:

Seismic failures:

- Most frequently reported: offsite power, electrical control panels, block walls, and interactions between buildings or systems
- Frequently reported: major building structures, switchgear, cable trays, fuel oil tanks, transformers, and pumps
- Also reported: switchgear chatter, ice condenser, AFW pipe, MFW heaters, containment fans, battery racks, invertors, battery chargers, accumulators, bus under voltage relays, motor control centers, electrical buses, surge tanks, control rod drive, and load centers, room cooling

Random failures:

- Most frequently reported: diesel generators
- Frequently reported: relief valves and AFW pumps

Operator failures:

- Most frequently reported: alignments and other actions to maintain AFW flow
- Frequently reported: actions to initiate cooling or recirculation
- Also reported: actions to reduce CCW heat loads, to cross-tie units, to shut down from the remote panel, to implement diesel procedures, and to reset relays

It is of interest to note that the SPRA IPEEEs assessed to date have indicated that the list of dominant contributors is not altered as a result of using different seismic hazard curves for seismic CDF quantification. That is, the dominant contributors are the same regardless of whether LLNL or EPRI hazard results [4,8] are used, and only minor changes in the ranking of dominant risk contributors have been observed.

Containment Performance Insights

Most containment performance insights were obtained based on qualitative assessments. However, a few of the IPEEEs assessed to date have employed a quantitative assessment of seismic containment performance. In some instances, the quantitative results are presented as frequencies of small and large radioactive releases, whereas, in other cases, they are presented in the form of frequencies of small and large containment failures. Some seismic PRA IPEEEs have also reported containment HCLPF capacities.

SMA IPEEEs assessed to date have generally implemented a qualitative, deterministic assessment of containment performance. Typically, the assessments have involved screening or walkdown examination of the following items:

- containment structural integrity
- containment penetrations, hatches, and seals
- containment cooling systems

No anomalous conditions have been reported with respect to containment structural integrity. In a few instances, outliers pertaining to containment penetrations and containment cooling have been identified.

Outliers, Plant Improvements, and Vulnerabilities

NRC guidelines for documentation of the IPEEE requested that licensees provide a definition of "vulnerability" for external events. Licensees have presented a variety of ways for defining plant vulnerabilities in the IPEEE submittals reviewed to date. In a few IPEEE submittals, the licensees have employed the guidelines proposed by NUMARC for vulnerability [9]. However, in most instances, no definition of vulnerability is proposed in the IPEEE submittals, and the submittal simply states that no vulnerabilities were found.

Many maintenance and minor improvements have been implemented as a result of the seismic IPEEEs. Some more significant plant changes have been made, based on analyses and resolution strategies implemented by the licensee. Some of the reported plant improvements would reduce seismic CDF, whereas others are simply undertaken to ensure proper plant maintenance.

Plant improvements related to seismic events have generally taken the form of various hardware fixes, maintenance actions, and maintenance procedural enhancements. Hardware fixes have included such items as: anchoring equipment, bolting cabinets together, improving existing anchorage or supports, installing missing fasteners and bolts, installing spacers on battery racks, eliminating potential interaction concerns, and replacing vulnerable relays. Maintenance actions have included the removal of corrosion on equipment anchorages, and application of corrosion protection. Maintenance and procedural (seismic housekeeping) enhancements have included provisions for proper storage of ladders, tools, gas cylinders, etc., and for proper parking of cranes and chain hoists. Similar types of improvements have been implemented with respect to seismic-fire interaction concerns. In addition, severe accident management guidelines have been considered for addressing some potential seismic scenarios.

Implications of Different PRA Methodologies

All of the seismic PRA IPEEEs assessed to date have generally followed the conventional seismic PRA methodology, such as described in NUREG/CR-2300 [10] and NUREG-1150 [11]. However, a hybrid variation on this methodology - the use of a surrogate element - has been employed in many seismic PRA IPEEEs. Table 1 indicates those IPEEEs for which the surrogate element has been employed.

The basis and approach for surrogate element modeling is discussed by Reed and Kennedy [12]. The overall concept of the surrogate element is to account, albeit approximately, for the effects of components that are screened out during the walkdown and screening phase of a SPRA. Hence, the potential failures of several components (that might normally be excluded from an SPRA model) are represented by the failure of a single surrogate element. Use of the surrogate element attempts to ensure that a potentially significant portion of the seismic CDF is not eliminated.

Based on the review findings reported in several IPEEE submittals assessed to date, it appears that the use of the surrogate elements in a SPRA may represent a reasonable alternate SPRA practice. However, the screening should be performed at a sufficiently high threshold, the capacity of the surrogate element should be assessed to be consistent with the screening threshold, and the surrogate element should be appropriately included in the plant logic model. Otherwise, the usefulness of this approach, and the validity of seismic PRA findings, may be compromised. This is revealed in some of the seismic PRA IPEEEs that used the surrogate element approach, in that the screening threshold was not chosen sufficiently high, the surrogate element was found to be a dominant risk contributor; thus, the true dominant contributors were "masked."

Implications of Different SMA Methodologies

The two different approaches to seismic margin assessment include the NRC methodology [13] and the EPRI methodology [6]. The principal insight from a comparison of application of these two seismic margin methodologies is that they provided substantially similar findings. It should be noted, though, that HCLPF capacities based on the EPRI method pertain to an 84th percentile non-exceedance probability (NEP), whereas those capacities based on a fragility approach are typically determined with respect to a 50th percentile NEP. Hence, if an NRC SMA is based on fragility calculations, the plant HCLPF should be adjusted to an 84th percentile NEP before making comparisons with HCLPF determinations from an EPRI SMA.

FIRE PERSPECTIVES

Methods

The analysis of internal fires has been a major aspect of virtually all of the IPEEEs reviewed to date, and each has included a substantial treatment of internal plant fires. Table 3 provides a summary listing of the plant submittals that have been reviewed to date. This includes identification of the method of analysis employed, the reported estimates of fire-induced CDF, the type of review performed, any plant improvements identified by the licensee in the fire portion of the submittal, and the fire areas identified as dominant contributors to the fire risk.

For the fire assessments, one of three methods has been employed; namely, (1) the EPRI Fire Induced Vulnerability Evaluation (FIVE) method [14], (2) fire PRA methods, or (3) a hybrid method that combines FIVE-based screening with detailed PRA-based quantification of unscreened scenarios. NUREG-1407 specifically identified both FIVE and fire PRA as acceptable approaches to analysis.

Methodological Differences

There are differences between the FIVE method and more traditional PRA approaches. For example, both FIVE and traditional fire PRA methods begin with screening assessments in which increasing levels of detail are introduced into each scenario through a progression of case evaluations. In FIVE a prescriptive sequence for this progression is set forth. In contrast, in a traditional PRA the progression is, to a large extent, tailored to each particular scenario under analysis. A second significant difference is that in a traditional PRA the screening analyses are always followed-up with detailed quantification of the unscreened scenarios. In contrast, the FIVE methodology itself stops at the end of the screening analysis. Another difference is that FIVE itself focuses attention primarily on Appendix R systems, Appendix R compliance, and on the availability of an undamaged redundant shutdown path without extensive consideration of the statistical reliability of those features. In contrast, a PRA will usually be based on a direct application of the internal events models; hence, a PRA would typically be expected to encompass a broader range of plant systems and components. However, in practice it is difficult to discern identifiable and systematic differences in the findings of an IPEEE as documented in the submittals based directly on the chosen method of analysis.

To date, none of the submittals reviewed can be characterized as a pure, direct, and complete application of the FIVE methodology. This includes the Palo Verde submittal which was performed in cooperation with EPRI as a demonstration application of FIVE. Virtually all of the FIVE-based submittals have included some modifications to the methodology. Some of these have been cited by the reviewers as having negatively impacted the quality of the studies. Even in the Palo Verde submittal, substantial modifications of the FIVE method were implemented.

One common example is that most licensees using the FIVE methods have substantially supplemented the Appendix R equipment list for analysis. FIVE does provide procedures for crediting non-Appendix R equipment, but this is primarily addressed in the context of the plant recovery analysis. Most FIVE-based IPEEEs have gone well beyond this practice and have given significant treatment to the risk impact of non-Appendix R systems. Many analyses based their equipment list on the IPE plant models. The extent to which the Appendix R equipment list was supplemented appears to vary from case to case, and has often been identified as a point of uncertainty in the reviews. RAIs to address this point have been forwarded to several licensees. While there is some uncertainty in this regard, it would appear that none of the submittals has been based on consideration of the Appendix R equipment only.

One difference that can be observed in the analysis results, as illustrated in Table 3, is that licensees that have applied the FIVE methodology have generally reported a higher "final" CDF as compared to those that have applied either the PRA or FIVE/PRA hybrid approaches. This can be largely attributed to the observation that the PRA methods refine scenarios further than FIVE; hence, more detailed quantification of equipment damage probabilities, scenario mitigation, and post-fire recovery are typically included in a PRA. The FIVE method itself stops at the level of scenario screening at which point a number of relatively conservative assumptions might still be incorporated into a given scenario. Hence, CDF estimates for a given scenario may be somewhat higher under the FIVE methodology. Other than this, it is difficult to discern significant differences in the results that can be directly attributed to the chosen method of analysis.

In certain cases some as yet unexplained results have been noted when comparing one IPEEE to another. In particular, plants that should be nominally similar from a risk and systems perspective have reported significantly different results both in terms of the absolute risk level and the dominant risk contributors even when the same basic methodology has been employed (e.g., FIVE). The current level of review has not allowed for an extensive exploration of such discrepancies. However, one likely possibility is that the variability in the analysis results is more a factor of the analyst than of the methods used.

The EPRI Fire PRA Guide

One area in which common methodological issues have been raised by reviewers is in the application by licensees of NSAC-181 [15] and the EPRI Fire PRA Guide [16]. These documents were intended to provide guidance to licensees on the performance of a fire risk assessment, but had not been reviewed nor approved by the USNRC for use in the IPEEE process. Nonetheless, these two documents have been cited extensively in several licensee submittals. This has led to the identification of a number of "generic" questions related to the guidance provided in these documents, and in particular, to that provided in the PRA Guide [17]. The following six items in particular are considered to be of broad applicability to fire risk assessments:

- use of electrical panel heat release rates that do not bound available test data,
- optimistic treatment of the timing and probability of main control room abandonment in the event of a control room electrical panel fire,
- optimistic values for the enclosure wall heat loss factor in fire modeling,
- inappropriate extrapolation of experimentally observed cable fire growth rates,
- optimistic treatment of manual suppression, and
- failure to consider event-specific performance shaping factors to modify internal events human error probabilities in the plant recovery analysis.

These items have been treated on a case-by-case basis for each submittal.

Walkdowns

One area specifically emphasized in NUREG-1407 is the importance of plant walkdowns to a fire risk assessment. Licensees had been asked to provide considerable detail of the walkdown process, participants, and findings. While much of this information has apparently been documented in second tier documents rather than in the submittals themselves, reviewers have generally concluded that the intent of this request has been met in that virtually all licensees have included plant fire walkdowns as a part of the fire assessment. Typically at least one walkdown has been performed together with the seismic analysis team to address seismic/fire interaction issues, and other walkdowns have been performed at various stages of the analysis as deemed necessary by the licensee. Reviewers have generally concluded that walkdown insights have had a significant impact on most of the assessments. In particular, most licensees have cited walkdowns as the basis for one or more of the following items: the identification of combustible loadings and ignition sources, development of input for fire modeling efforts, verification of fixed fire detection and suppression availability, verification of manual fire fighting access and equipment, verification of cable routings, and consideration of the Fire Risk Scoping Study issues.

Dominant Fire Risk Contributors

Results of the IPEEE fire assessments have generally been reported in terms of the risk-important fire areas. The PRA and FIVE/PRA hybrid studies typically identify those areas that represent the dominant contributors to fire risk along with risk quantification results. A typical submittal that used the FIVE methodology reported areas that survived the screening process along with a bounding estimate of fire risk.

Almost all licensees have included the main control room (MCR) in the list of fire risk-important areas. In only a few submittals has the MCR been screened from the analysis, and in such cases questions have been sent to licensees on their screening rational. The contribution from MCR fire generally includes scenarios both with and without MCR abandonment. In several submittals the MCR analysis used the methods and assumptions in the NSAC-181 and/or EPRI Fire PRA Guide for electrical panel fire heat release rates and MCR abandonment. For these cases MCR risk may have been underestimated, and the relative ranking of fires in the MCR as a risk contributor may be understated.

The second most commonly reported fire-risk important area is the cable spreading room (CSR). In only a few cases has the CSR been screened. In most of these cases questions have been sent to licensees on their screening rational. Two factors have been identified as especially significant to the CSR analysis. First, as expected, those plants with more than one cable spreading room (and hence substantial divisional separation) have estimated much lower CSR fire CDF values. Second, the presence or absence of fixed ignition sources other than cables (such as electrical panels or transformers) has significantly impacted the assumed CSR fire frequencies and the CSR fire CDF estimates.

The third area most commonly reported as fire risk-important are switchgear areas. Over half of the submittals reviewed to date have identified one or more switchgear areas as significant fire risk contributors. Several licensees have also identified either the turbine hall in general or selected areas of the turbine building as important to fire risk. Other areas that have been identified as important to fire risk on a plant-specific basis include battery and DC equipment rooms, cable penetration areas, switchyard areas, diesel generator rooms, the area housing the remote shutdown panel, areas associated with component cooling water systems, and various cable routing areas (e.g., corridors, chases, tunnels, and hallways).

Past fire PRA studies have commonly identified the cable spreading room, main control room and switchgear areas as dominant contributors to fire risk. In this regard, the results of the IPEEE fire analyses are nominally consistent with past PRA results and findings. Other areas have been identified as significant risk contributors on a case-by-case basis depending on the details of the plant layout, and in particular, the details of the plant cable routing paths. The finding that the turbine building was identified as an important contributor to fire risk for some plants was somewhat unexpected based on previous fire PRAs.

For most submittals reviewed to date only very limited information has been provided regarding fire-induced plant accident sequences. Indeed, for many submittals it has been difficult to discern which plant accident sequences have been considered in the analysis. In others, while the sequences considered were identified, their relative contribution to fire risk was not quantified. In some cases, all fires have been assumed to be bounded by a single accident sequence, for example a turbine trip or general plant transient with certain equipment rendered unavailable. In only a very few submittals has an explicit detailed discussion of fire-induced plant accident sequences been provided. Hence, it is not possible to reach a general conclusion regarding which plant accident sequences represent the dominant fire risk contributors.

Vulnerabilities and Plant Improvements

With the exception of Quad Cities, no licensees have reported any fire vulnerabilities (a further discussion of the Quad Cities submittal is provided immediately below). However, over half of the submittals reviewed to date have cited fire-related plant improvements. These are summarized in Table 3. In the fire area cited improvements have most commonly involved changes to plant procedures to address specific risk scenarios, and in particular, plant recovery actions and fire fighting procedures. A few plants have also identified physical changes to be made in the plant. For example, in one case, a licensee planned to move a particular set of cables out of a fire area in order to address a potential vulnerability of the remote shutdown station.

Quad Cities

The Quad Cities submittal is unique in several regards. First, the overall fire CDF estimate submitted by the licensee was higher than any other submittal reviewed to date. Quad Cities reported a fire CDF of about 5E-3 per reactor-year for each of the two units on site. In contrast, other licensees have reported fire CDFs ranging from 1E-9 to 2.2E-4 per reactor-year. Quad Cities is also the only licensee that has identified a potential fire vulnerability.

One of the major contributors to the identified fire vulnerabilities was the assumed reliability of the normal post-fire safe shutdown method. At Quad Cities, the safe shutdown path for certain postulated fire events relied on the utilization of equipment/systems from the non-fire affected unit. Further, the required operator actions were characterized as complex and required extensive operator manual actions and work-arounds. Due to the number and complex nature of these operator actions, a substantial failure probability was assumed; hence, recovery probabilities for a number of specific fire scenarios were assumed to be low.

Another unique aspect of the Quad Cities submittal is that many of the dominant fire scenarios were turbine hall scenarios. A total of seven individual turbine hall fire scenarios were each quantified as contributing in excess of 1E-4 CDF per reactor-year. This included lube oil fires from the three reactor feed pumps that together contributed over 30% of the overall fire CDF for each unit. This is somewhat unusual in comparison to past PRAs and to other IPEEE submittals. While it is widely recognized that the turbine hall does present a higher potential for large, uncontrolled fires than do other plant areas (due to the concentration of both ignition sources and high hazard fuels), past PRAs have generally found turbine halls to be minor fire risk contributors. This is because at most plants the critical safe shutdown systems are not located in the turbine hall. In the case of Quad Cities, many of the critical safe shutdown systems are either located in, or dependent on cables which pass through, the turbine hall. Hence, the Quad Cities results are attributable to unique design features of the plant. Additional factors that contributed to this result include:

- the critical cables were not assumed to be low-flame-spread (i.e., the cables were not IEEE-383 qualified); hence, they were assumed to be more vulnerable to fire spread and thermal damage, and

- many of the dominant Quad Cities scenarios involved postulated large oil spills that spread, ignite, and then develop so quickly that fixed suppression systems were assumed inadequate to prevent the critical damage.

As a part of the USNRC response to the Quad Cities submittal, an inspection team visited the plant in April of 1997 [18]. While this inspection was not undertaken as a part of the IPEEE review process, the team findings are of significant interest and will be factored into the IPEEE review. One important finding of the inspection team was that the licensee's results could not simply be attributed to gross conservatism in the analysis. Rather, while some sources of conservatism were clearly incorporated into the assessment, "the team does not view the Quad Cities IPEEE fire analysis as being overly conservative" [18].

The Quad Cities submittal is currently the focus of considerable attention by both the licensee and the USNRC. The licensee has committed to resolving the identified fire risk issues and has already taken measures to reduce the estimated CDF. Preliminary licensee proposals recently submitted for NRC review call for the addition of an independent safe shutdown capability for each unit. It is anticipated that substantial physical changes to the plant will be implemented to address the identified fire vulnerability.

HIGH WIND FLOOD AND OTHER EVENTS PERSPECTIVES

Methods

This section presents a summary of key findings from high winds, floods, and other external events (HFO) IPEEE submittals. NUREG-1407 recommends a progressive screening approach to identify potential vulnerabilities at nuclear power plants due to high winds, floods, and transportation and nearby facility accidents. This progressive screening approach can be summarized as follows:

(a) Demonstrate compliance with the 1975 Standard Review Plan (SRP).

(b) If the 1975 SRP criteria are not met, one or more of the following optional steps should be taken.
  - Determine if the hazard frequency is acceptably low. Demonstrate that the hazard frequency is less than 1.0E-05 per year.
  - Perform a bounding analysis. This analysis is intended to show that the hazard would not result in core damage frequency above the reporting criteria of 1.0E-06 per year.
  - Perform a probabilistic risk assessment.

All 41 IPEEE submittals listed in Table 4 addressed the possibility of HFO occurrence in those plants. The reviews of the HFO analyses of the IPEEE submittals were mainly based on the submittals. In some cases other documentation, such as a plant's final safety analysis report (FSAR) or a previous PRA, was consulted.

Table 4 provides a summary of the IPEEE HFO analyses. Typically, the licensees have been able to screen out most HFO initiators based on conformance with 1975 SRP criteria or by using quantitative analysis of the hazard frequency (denoted as "IE Screening" in Table 4). However, for some plants, PRAs or bounding analyses have been found to be necessary for addressing certain of the HFO events.

High Winds

Core Damage Frequency (CDF) Results:

Eight of the submittals have reported a core damage frequency from high winds and/or tornadoes ranging from 5.7E-05 to 3.7E-07 per year. Hurricanes, tornadoes or tornado-generated missiles lead to a loss of offsite power (LOSP). In all but one submittal LOSP is assumed to be unrecoverable. Typically, random failure of emergency AC power is the dominant sequence. Other random failure modes reported as leading to core damage are loss of service water, auxiliary feedwater, feed-and-bleed cooling, and high pressure injection. A number of other items were identified at individual plants, and these are summarized in Table 4. In a few of the submittals, optimistic assumptions have been noted and, in response to RAIs, have led to a reappraisal of core damage frequency.

Screening Results:

For those plants which did not perform a PRA evaluation of high wind risk, approximately half reported compliance with the 1975 SRP while most of the rest performed quantitative screening on the hazard frequency. For those plants which utilized quantitative screening, most reported bounding CDF estimates of 1.0E-07 per year or less.

External Floods

Core Damage Frequency (CDF) Results:

Four of the submittals have reported CDF from external flooding ranging from 1.0E-05 to 2.1E-08 per year. Dam break, hurricane, or intense precipitation causing river-rise lead to a LOSP. In all submittals, LOSP is assumed to be unrecoverable. One licensee reported that only random failures given LOSP resulted in core damage. The other submittals listed additional flood-related damage, including loss of intake structure, diesel generator building, auxiliary building, turbine building, and diesel fuel oil transfer pumps.

Screening Results:

For those plants which did not perform a PRA evaluation of external flooding risk, approximately half performed quantitative screening while almost all of the rest reported compliance with the 1975 SRP. One plant employed qualitative screening for its external flooding analysis. It has been observed at some plants that, even though flood hazards were found to screen out, a flood level just a few inches (or less) below the failure-incipient level might have an annual rate of occurrence of one to two orders of magnitude greater than the hazard for the failure-incipient level. Given the large uncertainties in site-specific flood hazard curves, screening may have been premature in some cases.

Potential failures of upstream dams, leading to flooding at the site, were considered and screened in many submittals. However, generic dam failure data has been employed in all cases, and these data do not consider site-specific information such as dam type and vintage.

Transportation and Nearby Facility Accidents

None of the submittals reported risk above the 1.0E-06 per year screening threshold from transportation and nearby facility accidents. Slightly more than half the submittals performed quantitative screening or PRA to address their transportation and nearby facility risk analysis. Almost all other licensees reported compliance with the 1975 SRP.

Other HFO Events

Core Damage Frequency (CDF) Results:

One submittal reported a core damage frequency contribution from lightning of 8.0E-06 per year and from snow and ice of 6.7E-06 per year. Lightning was assumed to cause a LOSP with other random failures required to result in core damage. In the ice and snow analysis it was found that the screenwell house, service building, and primary auxiliary building did not have roof live load capacities much more than the 100-year return interval ground snow load. Critical equipment failures due to roof collapse combined with other random failures led to core damage. One plant considered the effects of volcanic activity (including soot and ash deposition) but was able to screen out the event.

Screening Results:

NUREG-1407 does not require any explicit evaluation of HFO events other than high winds, external flooding, and transportation and nearby facility accidents. Consequently, most submittals did not report an analysis of "other" HFO events. For the few submittals which investigated "other" HFO events, most screened these events based on standardized and recognized screening techniques. In a few submittals where risk results are reported, almost all "other" HFO events were found to have core damage frequencies much less than 1.0E-06 per year.

Walkdown Insights

Almost all IPEEEs have included a walkdown for HFO events, although the walkdown results and procedures are generally not described in any detail. Most submittals did not provide any information regarding either walkdown findings or walkdown team composition. In many cases, walkdowns were employed to confirm no significant changes since the operating license was issued and compliance with the 1975 SRP. Two licensees did not conduct a walkdown.

A few of the submittals have provided some details regarding walkdown findings. Plant improvements associated with high winds/tornadoes and external flooding are noted. Susceptible components and structures were found to include: emergency feedwater storage tank, emergency switchgear room, condensate storage tank, diesel-driven fire pump, fire water system, demineralized water system, turbine-driven auxiliary feedwater pump, main steam lines, main feedwater lines, atmospheric relief valves, diesel generator exhaust, turbine building, instrument air system, feedwater/condensate system, station service water traveling screens and screen wash pumps, control room, and diesel fuel oil transfer pumps. Flood-related walkdown findings noted by licensees include: identification of several previously unidentified paths for entry of flood-water into critical structures at one plant and the potential to exceed the design load for a roof due to ponding at another plant.

Outliers and Plant Improvements

As noted in Table 4, six of the 41 submittals reviewed to date identified HFO-related plant improvements. These typically addressed changes to existing procedures and severe accident management guidance, although three submittals committed to hardware improvements. Primarily, plant improvements have been proposed or implemented for tornadoes/high winds, external flooding, and ice and snow. Procedural enhancements include sandbagging, closing doors, welding doors, hooking up pumps, and creating new circuits to reduce the risk from flood. In two submittals, development of severe accident management guidance to reduce the risk of high winds is being considered. Hardware improvements cited by licensees in the IPEEEs have included plugging flood entry pathways and installing portable water pumps to mitigate flooding. Some submittals noted that hardware changes undertaken in response to the IPE analysis (e.g., adding diesel generators) have also reduced or eliminated the risk from HFO events.

In a few plants, the licensees have proposed flood-related countermeasures which the reviewers have considered as highly optimistic. For example, one licensee has taken credit for sandbagging up to a level of nine feet. Another submittal credited all equipment in the turbine building and auxiliary building below grade as being capable of operation while submerged. These assumptions were questioned in the RAI process.

Containment Performance Insights

No submittals identified any HFO-related containment performance insights.

Human Action Perspectives

In some of the submittals, operator recovery actions to mitigate the effects of HFO-induced plant transients were documented. A number of submittals identified operator recovery of off-site power or diesel generators given a tornado or high-wind-induced LOSP as an important human action. For a few IPEEE analyses, given an external flood, sandbagging or installation of stop logs were employed as mitigation.

Generic Issues and Unresolved Safety Issues

Most submittals only provide a limited discussion concerning generic issues and unresolved safety issues. In all cases, the licensees have concluded, in submittals which provide relevant discussion, that USI A- 45 is closed out. With the exception of one submittal, roof ponding due to intense local precipitation was found to be accommodated by the existing plant design. For that one submittal, roof ponding only affected the spent fuel pool. Therefore, GI-103 ("Probable Maximum Precipitation (PMP)") was considered closed out in all submittals where this issue was specifically addressed.

CONCLUSIONS

The 41 IPEEE submittals reviewed to date have revealed numerous valuable perspectives concerning severe accident behavior and the most likely accident sequences at these plants. Licensees have benefitted from their IPEEE effort, and many have proposed or already implemented plant-specific improvements that will reduce the overall frequency of core damage and enhance the safety of these nuclear power plants. The perspectives gained through the reviews of IPEEE submittals concerning the completeness, level of detail, and overall quality, will enhance NRC's capability to focus more closely on specific issues related to external events when appropriate. A summary of the perspectives for each of the seismic, fire, and HFO areas of the IPEEE is presented below.

Seismic

In the seismic area, a number of generic findings have been identified. For the purposes of this paper, generic findings are defined as those frequently observed among plants, whereas plant-unique findings are those that are limited to perhaps just a single plant. Clearly, both plant-unique and generic insights have been revealed from the seismic IPEEE submittals assessed to date. Those findings considered generic include the following items:

- For most licensees it can be concluded that the principal objectives of the IPEEE program have been met, and that the seismic IPEEE program has had some impact on improving plant safety.

- Of the 41 submittals reviewed to date, only one, Haddam Neck, has identified a seismic-related vulnerability (four specific items were identified by the licensee as "risk outliers").

- Most of the licensees did list one or more seismic-related plant improvements as a part of the IPEEE submittal. These have included both changes to plant seismic response procedures and hardware modifications (such as improved anchorage).

- The seismic IPEEE program has addressed a number of generic issues (GIs) and unresolved safety issues (USIs), including USI A-45 ("Decay Heat Removal Requirements"), and GI-131 ("Potential Interaction Involving the In-Core Flux Mapping System at Westinghouse Plants"). In general, the seismic evaluations of the IPEEE are capable of addressing USI A-45 without any special additional considerations.

- For most applicable plants, GI-131 had been addressed through earlier upgrades and analyses. Some IPEEEs evaluated the capability of the in- core flux mapping system for beyond design basis seismic loads consistent with the IPEEE review level earthquake.

- The most commonly reported dominant IPEEE seismic risk contributors appear to be nominally consistent with the findings of past PRAs.

- The seismic IPEEE program has resulted in improved appreciation of the potential and effects of relay chatter. At many plants, low- ruggedness relays have been identified, but not on a widespread basis (i.e., typically, a small number of such relays have been discovered). In some cases, low-ruggedness relays have been replaced, whereas in most cases, relay chatter was screened out based on a consequence assessment. In many cases chatter was deemed acceptable because it was assumed to be recoverable.

- Seismic IPEEE studies of containment performance have led to improved appreciation of the potential for failure of containment cooling and isolation (including effects of relay chatter), and to improved understanding of the seismic capability of containment penetrations. In a few cases, containment-related concerns or improvements have been identified with respect to containment cooling and isolation. In general, however, containment safeguard equipment have been found to be rugged, and the seismic capability of containment safeguard systems is typically controlled by the capacity of the support systems.

- The level of detail of treatment of seismically-induced fires and floods has varied significantly among the IPEEE submittals. The majority of the submittals have not addressed these issues comprehensively; however, this is not surprising given that the state-of-the-art in this area requires more definitive development. Common findings of the seismic-fire interaction evaluations that have been performed include: suppression equipment (e.g., tanks, bottles, extinguishers) may need better seismic restraint, and the operation of fire pumps may be compromised by failure of fuel oil supply, batteries, or relay chatter.

- It should be noted that a consistent spectral shape has not been employed in reporting fragilities and margin capacities for components or for plants. Hence, it would be misleading to compare capacities among PRA studies or between PRA versus SMA studies. Additionally, the spectral shapes employed in seismic PRA studies for many eastern U.S. plants have not been effective in demonstrating seismic margin beyond the design basis.

- In some submittals, the status of plant improvements is not clear or has not been reported. In a number of instances, the findings of a submittal assume an upgraded plant condition that has not yet been implemented; hence, the reported CDF estimates or margins are conditional on completion of the cited upgrades.

In addition to these generic findings, listed below are some plant-unique findings that have not been fully revealed from past seismic evaluation studies of U.S. commercial nuclear power plants:

- A few eastern U.S. plants present significant core damage frequencies from seismic events, exhibiting seismic CDF values near or higher than 1E-04 per reactor year, regardless of whether the EPRI or LLNL seismic hazard results are used for seismic CDF quantification.

- Estimates of seismic capacities (HCLPF values) for certain plants can be extremely low as illustrated in Tables 1 and 2.

- Cable trays have been reported as outliers or dominant risk contributors in several IPEEEs. This is contrary to past experiments and PRA findings and may be due to overly conservative assumptions on tray seismic capacity or the impact of tray failure.

- Soil failures were identified as an important seismic risk factor in at least one submittal.

- Unreinforced block walls were identified as a dominant risk contributor at one plant reviewed to date.

- Several plants have taken credit for recovery of loss of offsite power by the use of (non-seismically qualified) combustion turbines or black start diesels whose seismic capacity may be marginal.

Fire

In the area of fire risk, a number of generic perspectives have been gained as a result of the reviews completed to date. These include the following items:

- For most licensees it can be concluded that the principal objectives of the fire portions of the IPEEE program have been met, and that the fire IPEEE program has had some impact on improving plant safety.

- Of the 41 plant reviews in process to date, only one licensee, Quad Cities, has identified a potential fire vulnerability.

- More than half of the licensees have reported some plant modifications as a part of the IPEEE fire submittal. These have most often involved procedural modifications, although hardware changes have also been cited by a limited number of licensees.

- The IPEEE results have generally reinforced the findings of past fire PRAs that have concluded that fires can represent a significant contributor to overall plant risk. While there are outliers at each end of the spectrum, most licensees have reported fire risk estimates in the range of about 1E-6 to 2E-4 per reactor year. These results are nominally consistent with past PRAs.

- The dominant fire risk areas most commonly reported by licensees include the main control room (MCR), cable spreading room, and one or more switchgear rooms. Other frequently reported areas include all or selected parts of the turbine hall, battery and DC equipment rooms, diesel generator rooms, areas associated with component cooling water, and various cable routing areas.

- The level of detail provided in the IPEEE submittals has not been sufficient to draw broad conclusions regarding the specific plant accident sequences that contribute most to fire risk estimates.

- Many licensees have based parts of their analyses on guidance that the USNRC had not reviewed nor approved for use in the IPEEE process. This has led to common questions being raised in a number of areas including electrical panel fire heat release rates, MCR abandonment, heat loss factors in fire modeling, cable fire growth behavior, manual suppression, and plant recovery analysis.

HFOs

Generic perspectives gained as a result of the reviews completed to date include the following items:

- For most licensees it can be concluded that the principal objectives of the HFO portions of the IPEEE program have been met, and that the HFO IPEEE program has had some impact on improving plant safety.

- No HFO vulnerabilities have been identified in any of the submittals reviewed to date.

- Of the 41 submittals reviewed to date, seven have cited HFO-related plant improvements as a part of the IPEEE. The cited improvements include both changes to procedures and hardware modifications.

- For some plants, a greater appreciation of the potential risk impact of high winds/tornadoes and external flooding/dam breaks has resulted from the IPEEE program. For these plants, tornadoes/high winds and external flooding risk has been found to range from 5.7E-05 to 2.1E-08 per year. Even though conservative bounding analyses estimates for high wind and external flood core damage frequency are reported in a few of the submittals, the risks associated with high winds and external floods (particularly for those plants located along rivers) have been identified as recurring concerns (i.e., being of concern at multiple plants).

- Transportation and nearby facility accidents have been screened out in all evaluations.

- GI-103 and USI A-45 were found to be closed out by all licensees who provided specific discussions on these topics.

- It has been observed at some plants that, even though flood hazards were found to screen out, a flood level just a few inches (or less) below the failure-incipient level might have an annual rate of occurrence of one to two orders of magnitude greater than the hazard for the failure-incipient level. Given the large uncertainties in site-specific flood hazard curves, screening may have been premature in some cases. Potential failures of upstream dams, leading to flooding at the site, were considered and screened in many submittals. However, generic dam failure data has been employed in all cases which does not consider important site-specific information such as dam type and vintage.

- Many submittals have simply used the IPE conditional core damage probability (CCDP), given loss of offsite power and loss of service water, without modeling specific significant impacts of high winds or floods, and thus, in some cases have significantly underestimated the CDF for such events.

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Table 1. Seismic CDF and HCLPF Results from SPRAs					
Plant Name	Seismic CDF (per ry)		HCLPF (g)	Spectral Shape	Surrogate Element Used?
	EPRI/Other	LLNL			
Catawba	1.6×10^{-5}	--	--	Site-Spec. (Sequoyah)	No
Cook	3.2×10^{-6}	1.0×10^{-5}	0.25	1989 LLNL	No
Diablo	4.2×10^{-5}	--	0.67	Site-Spec. (LTSP)	No
Haddam	2.3×10^{-4}	1.5×10^{-4}	<0.05	1989 EPRI	Yes
Indian Point 2	1.1×10^{-5}	w/LLNL	Not Reported	EPRI UHS	Yes
Kewaunee	1.1×10^{-5}	1.3×10^{-5}	0.23	1989 LLNL	Yes
LaSalle	7.6×10^{-7}	--	--		No
McGuire	1.1×10^{-5}	--	--	NUREG/CR-0098	No
Millstone	9.1×10^{-6}				No
NMP-2	2.5×10^{-7}	1.2×10^{-6}	0.50	NUREG/CR-0098	Yes
Oyster Creek	3.6×10^{-6}	6.4×10^{-6}	Not Reported	EPRI	No
Palisades	--	8.9×10^{-6}	0.22	1993 LLNL	Yes
Pilgrim	5.8×10^{-5}	9.4×10^{-5}	0.25	1989 LLNL	Yes
Pt. Beach	1.4×10^{-5}	1.3×10^{-5}	0.16	1989 LLNL	Yes
San Onofre	1.7×10^{-5} w/site-specific hazard		approx. 0.67	Site Specific	No
Seabrook	1.2×10^{-5}				No
S. Texas	$<1 \times 10^{-7}$				No
WNP-2	2.1×10^{-5}	(Site specific)	0.5 (Submitted) 0.25 (Review)	UHS (Site Specific)	Yes

Table 2. HCLPF Results from Seismic Margin Assessments				
Plant Name	Selected Method	Seismic Category	HCLPF (g)	Spectral Shape
Brunswick	EPRI	Focused-scope	>0.3	NUREG/CR-0098 Soil
Callaway	EPRI	Focused-scope	>0.3	NUREG/CR-0098 Soil
Clinton	EPRI	Focused-scope	0.3	NUREG/CR-0098 Soil
Comanche Pk	EPRI	Reduced-scope	--	SSE, 0.12g, Rock
Duane Arnold	EPRI	Reduced-scope	--	DBE site specific
Farley	EPRI	Reduced-scope	--	Original design spectra
Ft. Calhoun	NRC	Focused-scope	0.25	NUREG/CR-0098 Soil
Grand Gulf	EPRI	Reduced-scope	--	Original design spectra
Limerick	EPRI	Focused-scope	--	SSE, 0.15g, Rock
Monticello	EPRI	Focused-scope but performed reduced-scope	between 0.12 and 0.30 g	Screen with 0.3g NUREG-0098. Capacity based on 0.12g.
NMP-2	EPRI	Focused-scope	0.50	NUREG/CR-0098 Rock
Palo Verde	EPRI	Full-scope	0.3	NUREG/CR-0098 Soil
Peach Bottom	EPRI	Focused-scope but performed reduced-scope	--	Site specific (Rock)
Quad Cities	EPRI	Focused-scope	0.09	NUREG/CR-0098 Rock
River Bend	EPRI	Reduced-scope	--	Reg. Guide 1.60
Robinson	EPRI	Full-scope	0.28	NUREG/CR-0098 Soil
St. Lucie	Site-specific	Reduced-scope	--	SSE, 0.10g, Fill
Sequoyah	EPRI	Full-scope	0.27	NUREG/CR-0098
Shearon Harris	EPRI	Focused-scope	0.3	NUREG/CR-0098 Rock
Susquehanna	EPRI	Focused-scope	0.21	NUREG/CR-0098 Rock, Soil
Turkey Point	Site-specific	Reduced-scope	--	SSE, 0.15g, Rock
Vogtle	EPRI	Focused-scope	0.3	NUREG/CR-0098 Soil
Waterford	EPRI	Reduced-scope	--	Consistent with Reg Guide 1.60 and 1.61
Wolf Creek	EPRI	Focused-scope but performed reduced-scope	0.2	Original design spectra

Table 3: Summary listing of reviewed plants and IPEEE fire analysis results as cited by licensees.

Plant	Analysis Method	Review Level	Estimated CDF	Plant Improvements Cited by the Licensee	Significant Fire Areas
Brunswick	FIVE/PRA Hybrid	Step 1	3.4×10^{-5}	To be determined for scenarios with core damage frequency $>10^{-6}$/ry	Control room and cable spreading room
Callaway	FIVE	Step 1	8.9×10^{-6}	None	Control room and two ESF switchgear rooms
Catawba	PRA	Step 1	4.7×10^{-6}	None	Control room, cable spreading room, and component cooling room
Clinton	PRA	Screening	3.3×10^{-6}	Cable rerouting for zone CB-5a (included in risk estimate)	Control room, switchgear rooms, CCW pump rooms
Comanche Peak	FIVE/PRA Hybrid	Step 1	2.1×10^{-5}	None	Control room
Cook	PRA	Step 1	3.8×10^{-6}	None	Control room, diesel generator rooms, ESW system rooms, 4kV switchgear rooms, an MCC room, a battery room, a general area within the auxiliary building, and an area within the turbine building.
Diablo Canyon	PRA	Step 1	2.7×10^{-5}	None	Control room and cable spreading room
Duane Arnold	FIVE	Screening	Not Quantified beyond screening at 1×10^{-6}	Add restraints or remove gas bottles; optimize river water system outages and pre-stage fire hoses; conversion of one fire suppression system to dry pipe	Emerg. switchgear rooms
Farley 1&2	FIVE/PRA Hybrid	Screening	1.6×10^{-4}	Procedural enhancements to address seal LOCAs	Control room, switchgear rooms, EPR area, service water pump room, CCW pump room, low-voltage switchyard, cable spreading room, selected areas of turbine building.
Fort Calhoun	PRA	Step 1	2.7×10^{-5}	Procedural modifications to reduce possibility of interfacing system LOCA and Implementation of "Severe Accident Management Guidelines"	Control room and east basement of the auxiliary building, turbine building, and an electrical penetration area.
Grand Gulf	FIVE/PRA Hybrid	Screening	8.8×10^{-6}	None	Control room, switchgear rooms, Aux. Bld corridors
Haddam Neck	FIVE/PRA Hybrid	Step 1	6.1×10^{-5}	Several procedure changes including improved control of transient combustibles; addition of more sprinkler heads; one cable reroute (charging pump or auxiliary lube oil pump).	Control room, switchgear room A, primary aux. bld., diesel generator room B
Indian Pt. 2	PRA	Step 1	1.8×10^{-5}	None	Control room, cable spreading room, switchgear room

Table 3: Summary listing of reviewed plants and IPEEE fire analysis results as cited by licensees.

Plant	Analysis Method	Review Level	Estimated CDF	Plant Improvements Cited by the Licensee	Significant Fire Areas
Kewaunee	FIVE/PRA Hybrid	Step 1	9.8×10^{-5}	None	Auxiliary feedwater pump rooms, cable spreading room, and diesel generator room.
La Salle	PRA (RMIEP)	Step 1	3.2×10^{-5}	None	Main control room, auxiliary equipment room, and essential switchgear areas.
Limerick	FIVE	Step 1	less than 1×10^{-6}	Procedural changes for transient combustibles control and fire barriers	12kV switchgear room, static converter room, and remote shutdown room
McGuire	PRA	Step 1	2.3×10^{-7}	None	Control room, cable spreading room, vital instrumentation and control area, and auxiliary shutdown panel.
Millstone-3	PRA	Step 1	4.9×10^{-6}	None	Charging pump and component cooling pump area, cable spreading room, and control room
Monticello	FIVE/PRA Hybrid	Step 1	7.8×10^{-6}	None	Control room, 931' elev. of turbine bld., feedwater pump room, cable spreading room, 4kV switchgear room, "Division II" area of the control bld.
Nine Mile Pt. Unit 2	FIVE/PRA Hybrid	Step 1	1×10^{-6}	None	Control room
Oyster Creek	FIVE	Screening	7.7×10^{-6}	Considering upgraded anchorages for CO_2 bottles and replacement of deluge valves; reviewing anchorage for demineralizer trailer to mitigate potential inadvertent actuation of transformer fire suppression system.	480 VAC switchgear rooms and cable spreading room.
Palisades	FIVE/PRA Hybrid	Step 1	2×10^{-4}	May upgrade fire protection program, and based on the results, may re-quantify IPEEE.	Main control room, cable spreading room, turbine building, spent fuel pool equipment room, and aux. Bldg. El. 590'
Palo Verde	FIVE/PRA Hybrid	Screening	8.7×10^{-5} *	Separation of Train A/B control room circuits, development of offsite power recovery procedures	Control Room. Emerg. Switchgear rooms, DC equipment rooms, containment EPRs, turbine hall lower level, cable corridor.

Table 3: Summary listing of reviewed plants and IPEEE fire analysis results as cited by licensees.

Plant	Analysis Method	Review Level	Estimated CDF	Plant Improvements Cited by the Licensee	Significant Fire Areas
Peach Bottom 2 and 3	FIVE	Step 1	Not reported	Procedural modifications for containment venting, CO2 spurious actuation, and transient comb. controls; fire barrier and separation upgrades for various areas; removal of mercury switches in fire protection system; added seismic restraints on CO2 tanks;	Control room, cable spreading room, 4kV switchgear rooms, two areas of turbine bld., portion of the reactor bld. (FIVE screening survivors only, no CDF estimates generated for any areas)
Pilgrim	FIVE/PRA Hybrid	Step 1	2.2×10^{-5}	None	Main control room, switchgear rooms, vital MG set room, RBCCW / TBCCW pump rooms, turbine building heater bay, and main transformer
Pt. Beach	FIVE	Step 1	5.1×10^{-5}	Two additional diesel generators, auxiliary feedwater system independence from non-vital switchgear, and control room evacuation procedure	Control room, cable spreading room, auxiliary feedwater pump room, gas turbine room, vital and non-vital switchgear rooms, diesel generator rooms, and monitor tank room
Quad Cities 1 and 2	FIVE/PRA Hybrid	Step 1	5×10^{-3} (per unit)	To be determined, a preliminary licensee proposal to install independent remote shutdown capability has been submitted for USNRC review	Turbine bld. ground floor, turbine bld. mezzanine, cable spreading room, aux. elec. equip. room, Unit 1 reactor bld. grnd. flr., "old computer room," MG set area, battery switchgear rooms (2), unit 1 battery charger room (all above have CDF contribution >1E-4, several additional areas with CDF between 1E-5 and 1E-4 identified)
River Bend	PRA	Screening	2.2×10^{-5}	Modification of certain plant procedures to highlight potential fire impacts and responses	Control room, Div. 1 standby switchgear room, MCR ventilation room, ACU west room, HPCS and HPCS hatch area, B-tunnel, Aux. bld. W. cresent area
Robinson	FIVE/PRA Hybrid	Step 1	2.2×10^{-4}	Implementation of "Severe Accident Management Guidelines"	Control room, battery room, emergency switchgear room, and a yard transformer.
St. Lucie, Unit 1 and 2	FIVE	Step 1	1.9×10^{-4} unit 1; 1.2×10^{-5} unit 2	Procedures to allow power cross-tie between units and ensure that a roll-up door iskept closed	Control room, cable spreading room, and a switchgear room
San Onofre	FIVE/PRA Hybrid	Step 1	1.6×10^{-5}	Administrative procedure changes (3 cited)	Relay room, switchgear rooms (2), penetration rooms (2), turbine building, diesel generators

Table 3: Summary listing of reviewed plants and IPEEE fire analysis results as cited by licensees.

Plant	Analysis Method	Review Level	Estimated CDF	Plant Improvements Cited by the Licensee	Significant Fire Areas
Seabrook	PRA	Step 1	1.2×10^{-5}	Procedural improvements, expansion of a water suppression system, fire detectors in additional areas.	Control room, auxiliary building, switchgear rooms, and turbine building
Sequoyah	FIVE	Step 1	1.6×10^{-5}	None	Auxiliary building HVAC room, essential raw cooling water (ERCW), 125V battery room, 6.9kV switchgear room, and turbine building
Shearon Harris	FIVE/PRA Hybrid	Screening	1.1×10^{-5}	Remote shutdown procedural change to verify PORV status	Control room, switchgear rooms (2)
South Texas Project	PRA	Step 1	5.1×10^{-7}	None	Control room
Susquehanna	PRA	Step 1	less that 1×10^{-9}	Splash guard on a few electrical cabinets, provisions for draining water from cable spreading room.	None
Turkey Point Units 3 and 4	FIVE	Step 1	less than 2×10^{-4} *	For the cable spreading room, water proof cabinets and install dry pipe, reaction suppression system.	Control room, cable spreading room, and intake cooling water structure.
Vogtle	PRA	Screening	1.0×10^{-5}	None	Control Room, switchgear rooms (2), cable spreading rooms (2), cable chases, electrical penetration area, train A "electrical mezanine," train b "electrical raceway room"
Waterford 3	FIVE/PRA Hybrid	Screening	7.0×10^{-6}	Considering modification to protect certain chilled water pump cables.	Control room, cable spreading room, H&V mechanical room, elec. pent. area A, switchgear room, EDG B, Turbine building.
WNP 2	FIVE/PRA Hybrid	Screening	1.7×10^{-5}	Addition of batter constraints, considering anchorage upgrades for MCC's, various procedural changes.	Control room, corridor in Turbine hall, battery rooms, misc. elec. equip. areas.
Wolf Creek	FIVE/PRA Hybrid	Step 1	7.6×10^{-6}	None cited, under consideration	Control room, train A and train B switchgear rooms (2), elev. 2026 and elev. 2000 of the aux. bld. general area, south EPR, reactor trip switchgear room (MG sets)

* Inferred from the submittal, since the licensee has not provided a total core damage frequency

Table 4 Description of Methodology and Results for HFO IPEEEs			
Plant(s)	**Initiator: Methodology**	**Core Damage Frequency**	**Plant Improvements Cited by Licensee**
Callaway Clinton Farley Limerick Nine Mile Point #2 Seabrook Shearon Harris St. Lucie Susquehanna Wolf Creek	All: Conformance with 1975 SRP	Not applicable	None
Brunswick	High winds and floods: PRA Others: Quantitative screening	High winds: 4.0×10^{-6} Floods: 1.5×10^{-7}	High winds: Development of severe accident management guidance for high winds is being considered.
Catawba	High winds: PRA Floods: Quantitative screening Others: Conformance with 1975 SRP and quantitative screening.	Tornadoes: 2.6×10^{-6}	None
Comanche Peak	High winds: PRA Floods and others: Quantitative screening	Tornadoes: 3.7×10^{-6}	None
Cook	All: Quantitative screening	Not Applicable	None
Diablo Canyon	All: PRA	All: $<1\times10^{-6}$	None
Duane Arnold	High winds: PRA Tornadoes: IE Screening Others: Conformance with 1975 SRP	Fragilities and frequencies may be non-conservative	None
Fort Calhoun	Floods: PRA High winds and others: Quantitative screening	Dam break: 7.0×10^{-6} Periodic flooding: 3.0×10^{-6}	Dam break: Modify procedures and stage four portable water pumps. Two conduits, one into the intake structure and the other into the auxiliary building, were plugged.
Grand Gulf	High winds& floods: IE Screening Others: Conformance with 1975 SRP	Not Applicable	None

Table 4 Description of Methodology and Results for HFO IPEEEs			
Plant(s)	**Initiator: Methodology**	**Core Damage Frequency**	**Plant Improvements Cited by Licensee**
Haddam Neck	High winds: PRA Flooding and others: Bounding analysis	High winds: 5.7×10^{-5} Floods: 5.0×10^{-6} Lightning: 8.0×10^{-6} Snow and ice: 6.7×10^{-6}	High winds: Prior tornado PRA identified the need for an air-cooled diesel generator; make arrangements with fuel oil supplier to deliver additional fuel within 24 hours. Flooding: Ensure that flood door can be installed in eight hours— procedure and sufficient inspections. Snow and ice: Generate snow and ice removal procedure.
Indian Point 2	High winds and Tornados: PRA Flooding: Screened based on original licensing analysis Others: Conformance to 1975 SRP	Tornados: 3.0×10^{-5}	None
Kewaunee	High winds and others: Quantitative screening Floods: Qualitative screening	Not applicable	None
LaSalle	High winds and Others: PRA bounding analysis Floods: Qualitative screening	Tornadoes: 3.7×10^{-7} Aircraft crash: 5.7×10^{-7}	Information not available
McGuire	High winds: PRA Floods and others: Quantitative screening	Tornados: 1.9×10^{-5}	None
Millstone	High winds and others: Qualitative screening Floods: Conformance with 1975 SRP	Not applicable	Information not available
Monticello	High winds & Tornados IE Screening Aircraft Crash: Limited PRA Flooding: Non-conformance with 1975 SRP but emergency measures planned to mitigate flooding	High Wind: $<1\times10^{-6}$ Aircraft: $<9\times10^{-7}$	None
Oyster Creek	High winds & Tornadoes: IE Screeing Special Note: Frequency estimation may be non-conservative	Not applicable	None
Palisades	All: Quantitative screening	Not applicable	None
Palo Verde	Tornado: Limited PRA Others: Conformance with 1975 PRA	Tornado: $<1\times10^{-6}$	None
Peach Bottom	High winds and Tornados: Limited PRA Flooding: Conformance with 1975 SRP Transportation Accidents Limited PRA	Tornado: $<1\times10^{-6}$	None

Table 4 Description of Methodology and Results for HFO IPEEEs			
Plant(s)	**Initiator: Methodology**	**Core Damage Frequency**	**Plant Improvements Cited by Licensee**
Pilgrim	Floods: Conformance with 1975 SRP High winds and others: Quantitative screening	Not applicable	None
Point Beach	High winds: PRA Floods and others: Quantitative screening	High winds: $<1\times10^{-6}$	Addition of two new diesel generators (motivated by the IPE findings) cited as having a beneficial effect on HFOs.
Quad Cities	Conformance with 1975 SRP (Floods do not meet SRP-early warning and emergency procedures mitigate risk)	None presented	None
River Bend	High Winds & Floods IE Screening Others: Conformance with 1975 SRP	Not Applicable	None
Robinson	High winds: Bounding analysis Floods and others: Quantitative screening Plant-specific hazard: Lake Robinson Dam	High winds: 8.0×10^{-6}	High winds: Development of severe accident management guidance for high winds is being considered.
San Onofre	High winds IE screening Tornados Limited PRA Flooding Qualitative Screening Rail, aircraft and Transportation Accidents: IE Screening	All: $<1\times10^{-6}$	None
Sequoyah	High winds: Quantitative screening Floods and others: Conformance with 1975 SRP	Not applicable	None
South Texas Project	High winds: Conformance with 1975 SRP Floods: PRA Others: Qualitative screening	Floods: 2.1×10^{-8}	None
Turkey Point	High winds and floods: Quantitative screening Others: Conformance with 1975 SRP	Not applicable	High winds: Reinforcement of the Unit 1 and 2 (fossil plant) stacks, enhancement of the "Natural Emergencies" procedure. Floods: Refurbishment of the flood wall.
Vogtle	Progressive screening including volcanoes	Not Applicable	None
Waterford	Flooding : IE Screening Others: Conformance with 1975 SRP	Not Applicable	Committed to provide portable pump to prevent flooding in cooling tower areas.
WNP 2	Progressive Screening including volcanoes	Not Applicable	None

RESEARCH NEEDS IN FIRE RISK ASSESSMENT

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Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

Abstract

This paper identifies and discusses a number of fire risk assessment areas where research appears to be needed to: a) provide a better understanding of the risk contribution due to fires in nuclear power plants, b) provide improved support of ongoing and anticipated activities regarding nuclear power plant fire protection, and c) develop improved methods and tools to support the previous two objectives. An analytical representation of the current fire risk assessment process, augmented by information from a variety of sources, is employed to systematically identify potential need areas. The results of this process are expected to play a major role in the development of a fire research program.

1. BACKGROUND AND OBJECTIVE

Since being prompted by the Browns Ferry fire of 1975, a number of nuclear power plant fire risk assessments (FRAs) have shown that fires can be significant contributors to plant risk. The most important scenarios identified in these analyses tend to involve the occurrence of relatively infrequent fires whose location and severity are such that critical sets of plant equipment are likely to be damaged by such a fire, if it occurs. These general conclusions regarding the potential magnitude and character of nuclear power plant fire risk appear to be consistent with empirical evidence, where serious fire-induced challenges to reactor core cooling are not common events but have occurred.

While there is little argument about the potential importance of fires, the magnitude of the fire risk and the specific measures needed to efficiently manage this risk are not as clear when considering individual plants. Table 1 summarizes the results from a sample of FRAs performed on U.S. nuclear power plants. The variability in the estimated fire risk and risk contributors is due not only to plant-specific variations in design and operation, but also to variations in the methods and data used in the studies. Uncertainties in the current state of knowledge concerning the initiation, growth, suppression, and plant impacts of fire-induced nuclear power plant accident scenarios all contribute to this latter category of variability; they have raised significant concerns regarding the usefulness of current FRA tools in supporting proposed plant changes and the development of a risk-informed, performance-based rule for nuclear power plant fire protection.

The objective of this paper is to discuss a number of FRA methods and data areas where improvements appear to be needed. The discussion is based on the authors' experience in FRA methods development, the performance of FRAs, and the review of these studies; insights from the US. Nuclear Regulatory Commission's (NRC) ongoing review of Individual Plant Examinations of External Events (IPEEEs); experiences from the NRC's current efforts regarding the development of a risk-informed, performance-based fire protection rule; the results of a recent NRC-sponsored review of fire research issues

Table 1 - A Partial List of Fire PRAs for U.S. Nuclear Plants (Not Including IPEEEs)

Plant	Sponsor	Date	Fire CDF (/yr)	Total CDF (/yr)	Important Contributors[a]
HTGR (design)	USDOE	1979	1.1E-5[b]	4.1E-5[b]	CSR (only the CSR was analyzed)
Zion 1/2	Utility	1981	4.6E-6	4.9E-5	Electrical equipment room, CSR
Big Rock Point	Utility	1981	2.3E-4	9.8E-4	Station power room, cable penetration area
Indian Point 2	Utility	1982	2.0E-4[c]	4.7E-4	Electrical tunnels, swgr room
Indian Point 3	Utility	1982	6.3E-5[c]	2.3E-4	Swgr room, electrical tunnel, CSR
Limerick	Utility	1983	2.3E-5	1.5-5[d]	Equip. rooms, swgr room, access area, MCR, CSR
Millstone 3	Utility	1983	4.8E-6	7.2E-5	MCR, instrument rack room, CSR
Seabrook	Utility	1983	1.7E-5	2.3E-4	MCR, CSR
Midland	Utility	1984	2.0E-5	3.1E-4	Swgr room
Oconee	Utility	1984	1.0E-5	2.5E-4	
TMI-1	Utility	1987	8.6E-5	5.5E-4	MCC area, swgr room, cabinet area
Sav. River K Rx	USDOE	1989	1.4E-7[e]	3.1E-4[e]	MCR, maint. area, cable shaft, DG rooms
S. Texas Project	Utility	1989	< 1.2E-6[f]	1.7E-4	MCR
Diablo Canyon 1/2	Utility	1990	2.9E-5	2.0E-4	CSR, MCR
Peach Bottom 2	USNRC	1990	2.0E-5	7.6E-5[g]	MCR, swgr rooms, CSR
Surry 1	USNRC	1990	1.1E-5	2.8E-5[g]	Swgr room, MCR, aux bldg, cable vault/tunnel
La Salle 2	USNRC	1993	3.2E-5	1.0E-4	MCR, swgr rooms, equip rooms, turbine bldg, cable shaft
Grand Gulf 1	USNRC	1994	< 1.0E-8[h]	6.7E-5[g,h]	No areas found to contribute
Surry 1	USNRC	1994	2.7E-4[h]	4.3E-4[g,h]	Swgr room, cable vault/tunnel, containment, MCR

a) Area contribution > 1% total fire CDF; contributing areas prioritized by contribution (most important first); MCR = main control room, CSR = cable spreading room
b) Frequency of core heatup
c) Prior to plant modifications identified by risk study
d) Internal events only
e) Frequency of severe core damage
f) Total contribution from external events
g) Seismic contribution calculated using EPRI seismicity curve
h) Midloop conditions; instantaneous CDF is presented

[1]; a review of other recent papers and reports on current issues in FRA (e.g.,[2,3]); feedback from the NRC's Advisory Committee on Reactor Safeguards (ACRS); and informal discussions with researchers from universities, industry, government, and international organizations. The paper presents work in progress and does not represent a final NRC consensus position on research need areas, let alone a prioritization of needs. However, it is expected that the issues presented in this paper will factor strongly in the development of the NRC's fire research program. We note that, because of limited resources, this fire research program will probably focus on a limited number of issues. Collaboration with industry and international organizations is needed to ensure broad coverage of potential concerns.

It should be cautioned that the technical issues raised in this paper do not necessarily prevent the use of FRA as a decision support tool. While they are imperfect tools, FRAs have led to a better understanding of fire risk. This paper simply identifies areas where additional improvements in FRA tools and in fire risk understanding could be useful to NRC's fire protection activities.

2. APPROACH

In order to systematically identify FRA areas where research is needed, it must be first recognized that the intended research has the following three general technical objectives:

1) The research should lead to an improved understanding of the risk contribution due to fires in nuclear power plants. This understanding covers both quantitative aspects (e.g., the magnitude of the overall fire risk) and qualitative aspects (e.g., the scenarios that tend to dominate fire risk).

2) The research should provide support for ongoing or anticipated NRC program office activities. Examples include the development of a risk-informed, performance-based fire protection rule; fire protection inspections; and review of proposals to change a plant's current licensing basis. The last should include an evaluation of the impact of the proposed changes on risk (including fire risk).

3) The research should lead to the development of improved FRA methods and tools (including data), where such improvements are needed to support the first two objectives. Improvements are needed not only enable the assessment of situations not covered by current FRA, but also to improve the analysts' and decision makers' confidence in the results of an FRA.

These three objectives imply a broad range of research needs. In order to ensure that the identification of research needs is reasonably complete, an augmented analytical approach is employed. This approach first involves a systematic examination of the current FRA process and methodology, and the identification of areas where the current state of knowledge is weak and/or controversial. Next, to help ensure that the list of identified needs is not too heavily dependent on a particular view of fire risk and that it is not exclusively focused on methodological issues, the list is then supplemented using a information from a variety of sources, as discussed later in this section.

2.1 Fire Risk Assessment Process

Fire risk assessment for commercial nuclear power plants, as it is performed today, is little changed from the analytical process described in Refs. 4 and 5 and used in the Zion and Indian Point studies some 15 years ago [6,7]. Weaknesses in the elements of the approach, i.e., the data and tools for specific portions of the analysis, have been identified and progressively addressed in a number of studies (e.g., [8,9]). Furthermore, a number of remaining weaknesses in these elements, e.g., in the treatment of fire phenomenology, are the subject of discussion and ongoing research, as discussed below. However, the basic structure of the analysis has remained relatively constant.

In a typical FRA, the core damage frequency contribution due to a given fire scenario (where, in this discussion, a fire scenario is defined by the location and burning characteristics of the initiating fire) can be decomposed into three components: the frequency of the fire scenario, the conditional probability of fire-induced damage to critical equipment given the fire, and the conditional probability of core damage given the specified equipment damage. Formally accounting for the possibility of different levels of equipment damage and different plant responses following fire initiation,

$$CDF = \sum_i \lambda_i \left(\sum_j p_{ed,j|i} \left(\sum_k p_{CD,k|i,j} \right) \right) \tag{1}$$

where λ_i is the frequency of fire scenario i, $p_{ed,j|i}$ is the conditional probability of damage to critical equipment set j given the occurrence of fire scenario i, and $p_{CD,k|i,j}$ is the conditional probability of core damage due to plant response scenario k given fire scenario i and damage to critical equipment set j. Note that the second term addresses the issues of fire growth, detection, suppression, and component damageability, and that the third term addresses the unavailability of equipment unaffected by the fire and/or operator failures.

The three-term decomposition of fire risk presented in Eq. (1) is not unique; alternate decompositions (often involving more terms) can be found in the literature. From the standpoint of this paper, however, it is useful because each of the three terms tend to be addressed differently in current FRAs. In particular, the fire frequencies are generally estimated using simple statistical models for fire occurrences, the likelihood of fire damage is estimated using combinations of deterministic and probabilistic models for the physical processes involved, and the likelihood of core damage is estimated using conventional probabilistic risk assessment systems models. These different analytical approaches imply different methods and tool development needs.

2.2 Additional Sources of Information

The use of Eq. (1) in the identification of research issues is both a strength and a weakness. Clearly, it provides a framework for systematically identifying FRA issues especially relevant to Objective #1 listed above. This helps to ensure completeness. On the other hand, being model based, it provides a particular view of fire risk. If it is not carefully exercised, issues not explicitly addressed or even emphasized by the model may not be identified. For example, current FRAs are focused on the possibility of thermal damage to plant equipment. Although the general framework of Eq. (1) also applies to alternate damage mechanisms, e.g., smoke damage and damage due to suppression activities, specific issues relevant to these mechanisms, e.g., the frequency-magnitude relationship for smoke, can easily be overlooked.

Another weakness with the use of Eq. (1) in identifying research issues is that such an approach tends to focus on methodological and data issues. It does not necessarily address the users' needs implied by Objective #2; these needs may be satisfied by the performance of technical assessments using the current state of the art.

A variety of information sources are used to supplement the list of issues identified using Eq. (1). Formal sources include a recent NRC-sponsored review of fire research issues [1]; recent papers and reports on current issues in FRA (e.g., [2,3]); and feedback from the NRC's Advisory Committee on Reactor Safeguards. Informal sources include the authors' participation in the review of IPEEE studies and in NRC's current efforts to develop a risk-informed, performance-based fire protection rule; as well as informal discussions with researchers from universities, industry, government, and international organizations.

An important example of users' needs input is provided by Table 2. This table contains a list of 13 potential safety issues recommended for further study by the NRC staff. Twelve of these issues were identified as part of the NRC's fire protection rulemaking planning process [10]; the thirteenth issue (availability of safe shutdown equipment) was identified following the staff's review of the Quad Cities Individual Plant Examination of External Events (IPEEE) study [11]. Examination of these issues shows that a number of them (e.g., availability of safe shutdown equipment) can probably be addressed without additional methodological developments. However, they remain as potential research issues because their generic risk significance is not completely understood.

Table 2 - Supplemented List of Fire Protection Issues Identified in the Fire Protection Task Action Plan

Fire Impact on Reactor Safety
Availability of Safe Shutdown Equipment
Hot Shorts Resulting in Spurious Operations or Component Damage
Control Room/Cable Spreading Room Interaction with Remote Shutdown Capability
Smoke Effects on Personnel/Equipment
Explosive Electrical Faults
Compensatory Measures for Fire Protection Deficiencies
Seismic Fire Interactions
Fires During Non-Power Operations
Broken/Leaking Flammable Gas Lines
Reliability of Fire Barriers
Equipment Protection from Fire Suppression System Actuation
Fire Detection Methods

3. POTENTIAL RESEARCH NEEDS

Table 3 presents a list of the potential fire research issues identified using the approach described above. As indicated by the note at the bottom of the table, most of these issues are grouped according to the general FRA area of analysis (e.g., fire initiation analysis). The remainder of the issues deal with either: a) problem-specific, integrated treatments of fire initiation, equipment damage, and plant response, or b) issues not directly derived from the FRA analysis process. Table 4 groups these issues into topic areas, where topic areas can be distinguished by the general type of analysis (e.g., statistical vs. phenomenological) as well as subject matter. Note that the orderings of the potential issues and topics are not based on any notion of relative importance. Discussions within the NRC regarding issue and topic prioritization are ongoing.

The remainder of this section provides background information relevant to the issues listed in Tables 3 and 4.

3.1 Fire Initiation

According to a recent NRC study, the frequencies of fires in key U.S. nuclear power plant compartments have not changed dramatically when comparing the periods 1965-1985 and 1986-1994 [12].

Table 3 - Potential Research Issues

Issue ID	Issue Title	Issue ID	Issue Title
I1	Adequacy of fire events database	P1	Circuit interactions
I2	Scenario frequencies	P2	Availability of safe shutdown equipment
I3	Effect of plant operations, including. compensatory measures	P3	Fire scenario cognitive impact
I4	Likelihood of severe fires	P4	Impact of fire induced environment on operators
E1	Source fire modeling	P5	Role of fire brigade in plant response
E2	Compartment fire modeling	R1	Main control room fires
E3	Multi-compartment fire modeling	R2	Turbine building fires
E4	Smoke generation and transport modeling	R3	Containment fires
H1	Circuit failure mode and likelihood	R4	Seismic/fire interactions
H2	Thermal fragilities	R5	Multiple unit interactions
H3	Smoke fragilities	R6	Non-power and degraded conditions
H4	Suppressant-related fragilities	R7	Decommissioning and decontamination
B1	Adequacy of data for active and passive barriers	R8	Fire-induced non-reactor radiological releases
B2	Barrier performance analysis tools	R9	Flammable gas lines
B3	Barrier qualification	R10	Scenario dynamics
B4	Penetration seals	R11	Precursor analysis methods
S1	Adequacy of detection time data	R12	Uncertainty analysis
S2	Fire protection system reliability/availability	O1	Learning from experience
S3	Suppression effectiveness (automatic, manual)	O2	Learning from others
S4	Effect of compensatory measures on suppression	O3	Comparison of methodologies
S5	Scenario-specific detection and suppression analysis	O4	Standardization of methods

Note: The first character in the issue IDs refers to the type of issue. I = fire initiation, E = fire-induced environment, H = hardware impact, B = fire barrier, S = fire detection and suppression, P = plant response, R = integrated fire risk, O = other.

Table 4 - Potential Fire Research Issues Grouped By Topic Area

Topic ID	Topic Title	Issue ID	Issue Description
T1	Fire events database	I1	Adequacy of fire events database
T2	Fire initiation analysis	I2	Scenario frequencies
		I3	Effect of plant operations, incl. compensatory measures
		I4	Likelihood of severe fires
T3	Fire modeling toolbox: assessment and development	E1	Source fire modeling
		E2	Compartment fire modeling
		E3	Multi-compartment fire modeling
		E4	Smoke generation and transport modeling
		H2	Thermal fragilities
		H3	Smoke fragilities
		H4	Suppressant-related fragilities
		R12	Uncertainty analysis
T4	Fire barrier reliability analysis	B1	Penetration seals
T5	Fire barrier qualification and thermal analysis	B2	Adequacy of data for active and passive barriers
		B3	Barrier performance analysis tools
		B4	Barrier qualification
T6	Detection and suppression analysis	S1	Adequacy of detection time data
		S2	Fire protection system reliability/availability
		S3	Suppression effectiveness (automatic, manual)
		S4	Effect of compensatory measures on suppression
		S5	Scenario-specific detection and suppression analysis
T7	Circuit failure mode and likelihood	H1	Circuit failure mode and likelihood
T8	Impact of fires on operator performance	P3	Fire scenario cognitive impact
		P4	Impact of fire induced environment on operators
		P5	Role of fire brigade in plant response
		R10	Scenario dynamics
T9	Risk significance of main control room fires	P1	Circuit interactions
		R1	Main control room fires
T10	Risk significance of turbine building fires	R2	Turbine building fires
T11	Risk significance of containment fires	R3	Containment fires

Table 4 - Potential Fire Research Issues Grouped By Topic Area (continued)

Topic ID	Topic Title	Issue ID	Issue Description
T12	Fire PRA applications issues	P2	Availability of safe shutdown equipment
		R4	Seismic/fire interactions
		R5	Multiple unit interactions
		R6	Non-power and degraded conditions
		R9	Flammable gas lines
		O3	Comparison of methodologies
T13	Non-core damage issues in fire risk assessment	R7	Decommissioning and decontamination
		R8	Fire-induced non-reactor radiological releases
T14	Precursor analysis methods	R11	Precursor analysis methods
T15	Experience from major fires	O1	Learning from experience
T16	International cooperation	O2	Learning from others
T17	Fire PRA guidance and standardization	O4	Standardization of methods

The computed reductions (in most cases) and increases (in the case of the turbine building) are generally not large when considering: a) the uncertainties in the estimated frequencies, and b) variability in reporting practices. Other than addressing the need for a maintained database, therefore, it may appear that little methodological work needs to be done in this area. However, a closer examination of the way in which empirical fire frequencies are employed in FRAs reveals some issues that need to be addressed.

First, and most obvious to FRA practitioners and reviewers, is the reduction of fire frequencies performed in most detailed FRAs to accommodate the fact that not all fires are risk significant, i.e., that a fire must have the proper location and severity characteristics to be a potentially important cause of critical equipment damage. In a number of FRAs, "location fractions" are employed to reduce plant area-based fire frequencies to account for geometrical factors; other FRAs use plant component-based fire frequencies for this same purpose. Regarding fire severity, "severity fractions" are widely used to address the fraction of fires (in a given compartment or involving a given component) that have the potential to cause significant damage in a relatively short amount of time.

Current reduction factors used to address location and severity considerations can reduce the compartment fire frequencies (the λ_i) by one or more orders of magnitude. However, the basis for these reduction factors is not strong. Early studies relied heavily on analyst judgment. Attempts to reduce the influence of judgment have led to: a) the component-based approach to fire frequency, employed in the Electric Power Research Institute (EPRI) Fire-Induced Vulnerability Evaluation (FIVE) methodology [13], and b) event-based estimation of severity fractions (e.g., [14-16]). However, these approaches are not without problems. Regarding the location issue, the FIVE approach requires an assumption that the total frequency of fires involving a specific class of equipment is constant from plant to plant. (Note that relaxation of this assumption will require an estimate of the population base, including non-safety as well as safety equipment.) This assumption neglects differences in the effectiveness of fire prevention programs, but is not, in general, expected to have a major impact on fire frequency estimates.

The concerns with the event-based treatment of the severity issue are potentially more significant. These include: ambiguity in the data (qualitative event narratives are used to determine if a given fire was severe); possible double-counting of the impact of suppression in the data (effective suppression may be the reason why a particular fire was not reported as being severe, but fire suppression is modeled separately in the FRA -- see Section 3.2.4); neglect of possibly significant differences between conditions (e.g., fuel bed geometry) of the event and those of the situation being analyzed in the FRA which can affect the severity of the fire; and scarcity of data for the large, transient-fueled fires that have been predicted to dominate fire risk in a number of studies.

Other issues related to the estimation of fire frequencies include: the effect of plant operations on fire frequency, the frequency of self-initiated cable fires, and the potential significance of unreported fires. Regarding the first issue, current analyses are unable to quantitatively predict the impact of such measures as the use of fire watches or the existence of administrative controls on the storage of transient combustibles on the frequency of fires, let alone the frequency of severe fires. This is an important problem from a fire risk management point of view, e.g., in situations where such compensatory measures as fire watches are proposed to account for temporary fire protection deficiencies. Regarding the second issue, tests have shown that electrical ignition of fires involving IEEE-383 rated cables is difficult (e.g., see Ref. 17). A practical FRA question is, for compartments containing only rated cables, what is the frequency of cable fires? Is it sufficiently low that the analysis only need consider transient-fueled fires? The third issue is related to the severity factor issue: many fires in U.S. nuclear power plants do not cause sufficient damage to meet reporting criteria. The fire frequencies used in FRAs, therefore, are based solely on reported fires. While it has been argued in past FRAs that only the reported fires are potentially risk significant and should be considered when estimating the λ_i, a detailed technical basis to support this argument has not been developed.

The preceding issues deal with the problem of quantifying the likelihood of fire occurrence. A related issue concerns the establishment of conditions for the next stage of the FRA, the estimation of the likelihood of equipment damage (see Section 3.2). Current methods for performing this next stage generally rely upon fire environment simulation models, and these models require the specification of the initial conditions for a given simulation. The problem is that current fire frequency analyses provide, at most, the frequency of "small" and "large" fires in a specified compartment or involving a specified component. They do not provide the physical characteristics associated with these "small" and "large" fires needed by the simulation models. This ambiguous interface between the fire frequency and equipment damage analyses allows significant analyst discretion. For example, the Indian Point study [7] assumes that "large" fires have a severity equivalent to a 2-foot diameter oil fire, while the Surry NUREG-1150 study [18] assumes that this is the equivalent severity of "small" fires. In the Quad Cities IPEEE [11], all main feed pump fires are analyzed as if they involve the release of a pump's entire lube oil inventory into a diked sump area and subsequent ignition of the oil; there is no distinction between large and small fires.

Fire frequencies have, to date, been treated as empirical parameters which can be directly estimated from data. The issues discussed above show that this treatment may need to be re-examined, especially if FRA is to play a stronger role in risk management. As argued in Ref. 19, a more mechanistic, systems modeling approach which specifically addresses the possible scenarios leading to fire ignition and the different outcomes of these scenarios, and does so within the constraint of available data, appears to be needed.

3.2 Equipment Damage

Given a fire in a nuclear power plant compartment, the conditional probability of damage to key equipment needs to be determined. In a detailed FRA, the assessment typically involves a prediction of the fire-induced environmental conditions, an assessment of the likelihood of equipment damage under these conditions, and an assessment of the likelihood that the fire will not be detected and suppressed before equipment damage occurs. The analysis may also consider the effectiveness of fire barriers in preventing fire damage to protected equipment and in preventing fire growth to neighboring compartments.

3.2.1 Fire Environment

Characterization of the fire-induced thermal environment for the purposes of probabilistic risk assessment (PRA) requires the estimation of the time-dependent temperature and heat fluxes in the neighborhood of the safety equipment of interest (i.e., the "targets"). This requires the treatment of a variety of phenomena as the fire grows in size and severity, including the spread of fire over the initiating component (or fuel bed), the characteristics of the fire plume and ceiling jet, the spread of the fire to non-contiguous components, the development of a hot gas layer, and the propagation of the hot gas layer or fire to neighboring compartments. It also requires an appropriate treatment of uncertainties in the structure and parameters of the models used to perform the analysis.

It is well recognized in the fire sciences community that there are limitations in our current ability to model fire behavior (e.g., see [20]). Even current "field models" (numerical computational fluid dynamics simulation models) adapted to fire applications do not address all of these limitations as they deal with fluid flow and heat transfer but not with fundamental combustion processes. The development of a detailed level understanding of fire phenomenology is a long term prospect. Given the risk assessment perspective that near term decisions must be supported with the best information presently available, the question is if the tools available are "good enough." More precisely, are there tools to treat all fire scenarios of interest, are the limitations of these tools known, and are the biases and uncertainties in their predictions understood?

A fire scenario involves the development of a specified source fire over time. Three source fires of special interest in nuclear power plant FRAs are cable tray fires, electrical cabinet fires, and very large oil fires. The risk significance of cable tray fires and electrical cabinet fires, has long been recognized. More recently, very large oil fires have been found to be important in situations where severe turbine building fires can significantly impact efforts to achieve safe shutdown (e.g., see Refs. 11 and 21). As discussed below, there are considerable uncertainties in key parameter values characterizing cable and cabinet fires. On the other hand, while the physical properties of oil are reasonably well understood, the ability of current FRA models to accurately predict the behavior of very large oil fires under realistic plant conditions is of concern, due to such complications as flame obstructions and oxygen starvation (both local and global).

Given a source fire, the next questions to be answered by the thermal environment analysis involve fire growth within the compartment and spread to neighboring compartments (neglecting for the moment the effect of fire suppression activities). Characteristics that can affect these processes include the compartment geometry and ventilation, location of the source fire, and, in the case of the multi-compartment fires, the effectiveness of barriers (see Section 3.2.3). As will now be discussed, these characteristics are not completely addressed by the models currently used in FRA.

To date, U.S. nuclear power plant FRAs have used quite simple zone model-based tools, e.g., the correlations provided as part of the EPRI FIVE methodology [13] and the COMPBRN computer code [22,23], to predict the thermal environment due to a variety of fire sources, including cable tray, electrical cabinet, and oil pool fires. However, it is not always recognized in FRAs that these tools have been developed to address specific classes of fire problems and are not applicable to all situations. For example, the inherent zone modeling assumptions in both FIVE and COMPBRN do not address many practical complexities (e.g., obstructions in the fire plume, complex compartment geometry, complexities in forced ventilation flow, physical movement of fuel, room flashover) which can be important in some analyses. Further, the correlations employed implicitly or explicitly by these models are not appropriate for all situations. Some scenarios of potential concern include very small fires (e.g., single wire electrical insulation fires), very large fires (e.g., very large oil spill fires), or elevated fires. Unfortunately, the limitations of these simple models have not been succinctly characterized to inform FRA analysts, many of whom may not have strong background in fire science, when they should be wary of the model predictions.

Even in cases where the models are appropriate, the uncertainties in their predictions have not been completely characterized. These uncertainties stem from two sources: the uncertainties in model input parameters, and the uncertainties in the fire models themselves.

Regarding the first source of uncertainty, all compartment fire models require, as input, information concerning the burning characteristics of the fire and the physical characteristics of the compartment. The latter can usually be specified with relatively low uncertainty. However, this is not typically the case with the former. Whether the fire model requires a time-dependent heat release rate, as is the case with many widely available zone models (e.g., CFAST [24]), or more detailed information such as mass pyrolysis rates per unit fuel area and radiation feedback coefficients, as is the case with COMPBRN, the data available to estimate the required parameters are often sparse, especially in the case of cable fires and electrical cabinet fires. Further, the data may be sufficiently ambiguous that their applicability to a particular FRA scenario is uncertain. This problem has led to a controversy in the treatment of heat release rate data for electrical cabinet fires in recent IPEEEs [3].

In the relatively small number of FRAs where parameter uncertainties have been formally propagated through a fire model, the probability distributions used to quantify the uncertainties in these parameters are relatively broad. It should be noted, however, that even these broad distributions do not necessarily reflect possible biases resulting from differences between the manner in which experimental measurements are made (e.g., using bare thermocouples above cable jackets) and the manner in which they are used in the FRA (e.g., as cable surface temperatures). Because of the data sparseness, near term efforts are needed to ensure that all relevant information is readily available for use in FRA. Because of the possible biases, efforts are also needed to ensure that this information is properly used. Formal Bayesian techniques for quantifying uncertainty may be required (e.g., see Ref. 25). Longer term efforts to increase the amount of quality data may also be needed.

Regarding the second source of uncertainty, it has already been pointed out that current fire models are highly approximate. Furthermore, benchmarking calculations of direct relevance to nuclear power plant FRA have been extremely limited. Consequently, there are significant uncertainties in the model predictions even in situations where the model input parameters are known quite well. (Note that the uncertainties in the input parameters complicate the assessment of the models' predictive capabilities [26].) The problem is that the issue of model uncertainty, which was considered in a preliminary fashion in early FRAs (e.g., [6,7]), has not been seriously addressed in more recent studies. This is partially due to the fact that the risk assessment community has not reached a consensus on how to treat model uncertainty (see Ref. 27). Another reason is that the data needed to quantify uncertainty in fire model predictions, regardless of approach, are limited. (Note that, as pointed out by Ref. 1, not all of these data are currently available to analysts.) Consequently, the uncertainties in FRA fire model predictions, even for such widely used variables as the average hot gas layer temperature, are not well known for most situations of interest. There is a clear near term need to characterize these uncertainties, making the best possible use of available (and potentially available) information in this process. As in the case of input parameter uncertainties, longer term efforts to generate more benchmarking data may also be needed.

The above discussion has focused on the prediction of the thermal environment induced by a fire. Predictions of non-thermal environmental characteristics due to the fire (e.g., smoke) or efforts to put the fire out (e.g., humidity) have historically received far less scrutiny in nuclear power plant FRAs. However, with the increasing use of sensitive electronic components in advanced instrumentation and control systems, and with increasing concern of the environmental impacts of fires on operator performance, these issues are gaining increased attention. Models such as CFAST are capable of predicting the buildup of smoke within a room and the transport of smoke to other rooms. However, research efforts generating the basic data needed to estimate smoke generation rates characteristic of nuclear power plant fires and to quantify the uncertainties in these rates are still in their early stages (e.g., see [28]). Also, as in the case of the thermal environment models, the uncertainties in the smoke buildup and transport models need to be assessed.

In summary, there appears to be a short term need to: define the limitations of the fire models used (or proposed for use) to treat fire scenarios of interest in FRA, improve the characterization of uncertainties in the input parameters for these models, and improve the characterization of uncertainties in the models themselves. Possible longer term needs include: additional data for input parameter and model uncertainty quantification, and improved fire models to address key limitations in current models (again with respect to the scenarios of interest in FRA).

3.2.2 Hardware Performance

Given a predicted environment for a piece of equipment, the FRA needs to determine the likelihood of equipment failure and the mode of failure. Because of the common cause failure potential of cable fires, the key concern is the fragility of electrical cables. However, the fragilities of other potentially vulnerable equipment, e.g., electro-mechanical and electronic components in electrical cabinets, are also of interest. In principle, the multiple threats posed by heat, smoke, and fire suppressants may need to be addressed. In practice, only the effects of heat have been treated in mechanistic analyses.

Current FRA treatments of equipment failure due to heat are very simple; it is generally assumed that damage will occur if a representative temperature (e.g., the surface temperature of a cable) exceeds a threshold value. In some analyses, component damage is also assumed if the incident heat flux exceeds a critical value. When component temperature criteria are used, conservative approaches (e.g., assuming the component is at the local environment temperature) or simple heat transfer models (e.g., lumped capacitance models or one-dimensional transient heat conduction models in the case of cables) are employed.

Similar to predictions of the fire-induced environment, predictions of thermal damage are subject to uncertainties and biases in both parameters (e.g., the cable damage temperature) and models. Potentially important biases include neglect of the difference between the cable surface temperature and its temperature in the vicinity of the conductors and the neglect of possible phase changes. The material properties of key equipment, especially electrical cables, and the potential effect of improved modeling (e.g., to determine the temperature of equipment in electrical cabinets) need to be better understood.

Current FRAs do not explicitly address the issue of smoke damage. (It can be argued that smoke damage is partially addressed in scoping analyses which assume that any fire within a given plant area damages all equipment in that area. Such an approach, of course, does not cover smoke-induced damage in neighboring areas.) A number of studies have been performed or are being performed to investigate the impact of smoke on electronics (e.g., [28,29]). However, the effect of smoke on the reliability of other types of potentially vulnerable equipment (e.g., switchgear) is not currently being studied and may need to be addressed.

Regarding the failure of equipment due to the application (or misapplication) of suppression agents, an analysis of the potential risk significance of this issue has been performed [30]. This analysis employs historical information on suppression system actuations and equipment failures to estimate generic equipment fragility. It is not clear how much of the uncertainty in equipment response is due to variations in equipment layout (with respect to the suppression system), how much is due to variations in equipment design, and how much is due to other factors (e.g., room ventilation, duration of exposure). A more detailed investigation of suppressant-related equipment fragility may be required, especially for the seismic-fire interactions scenarios determined to be potentially important by Ref. 30.

Besides determining the likelihood of equipment failure , the FRA needs to specify the failure mode, i.e., how the failure occurs. Of particular interest when dealing with electrical control or power cables are circuit failures that lead to loss of function and those that can lead to spurious actuation of plant equipment. The latter failure mode, typically referred to as "hot shorts" in FRAs, has been shown to be an important and sometimes even dominant contributor to risk. In such cases, the scenarios often involve the spurious opening of one or more valves in the primary system boundary and a subsequent loss of coolant accident (LOCA).

From an FRA methods standpoint, the concern is that hot short analyses are generally simplistic. The probability of a single hot short is commonly based on a generic probability distribution derived subjectively in 1981 from a limited amount of information [31]. (The distribution, assumed to be lognormal, has a 5th percentile of 0.01 and a 95th percentile of 0.20; its mean value is 0.07.) The probability of multiple hot shorts is typically obtained by multiplying this probability an appropriate number of times. The latter procedure ignores the potentially significant impact of state-of-knowledge dependencies. More importantly, both it and the original single hot short distribution do not reflect such presumably important issues as the circuit design, the function of the cable, and the characteristics of other cables in the vicinity.

Given the reported risk significance of hot short scenarios, there is a clear need for improved models and data for estimating the likelihood of fire-induced spurious actuations. It should be noted that the importance of analyzing different circuit failure modes will probably increase when the effects of fire on instrumentation, which are generally not treated in current FRAs, are addressed.

3.2.3 Fire Containment

As part of determining the immediate environment of equipment potentially affected by a fire, the FRA needs to consider the effectiveness of fire barriers.[1] The question is, from an FRA perspective, the degree to which the barrier reduces the likelihood of damage to protected equipment.

Current FRAs treat barriers fairly simply and sometimes simplistically. For barriers separating fire areas, many FRAs neglect the possibility of barrier failure. Others that treat this possibility use generic failure probabilities reported in a number of NRC FRAs (e.g., Ref. 9). We note that the data used to estimate the failure probabilities have not undergone extensive review, and, further, that they have been widely misinterpreted. In the original analysis of "barrier failure rates," the total number of observed barrier

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[1]Note that the often-quoted fire duration ratings of fire barriers (e.g., as determined by the ASTM E-119 furnace test) should be taken as relative indications of barrier effectiveness. The fire sciences community has agreed that the quantitative model relating fire loads and fire severity that underlies these ratings is obsolete [32]. This means, for example, that a 3-hour barrier will not necessarily prevent the spread of fires with an "equivalent severity" (as computed from the fire load) of less than 3 hours.

failures[2] is divided by an estimated exposure time. These failure rates have been quoted and used as failure probabilities. We also note that the original analysis is of limited scope and does not incorporate recent data. It appears that for scenarios where barrier reliability plays an important role, there is a need to establish a firmer basis for quantifying this reliability.

For barriers separating equipment within compartments, the barriers are usually either assumed to be 100% reliable or are entirely neglected. Even when physical models for barrier performance are employed (e.g., COMPBRN provides a one-dimensional steady state heat conduction model), these models do not address such behaviors as gross distortion and mechanical failure of the barrier system. Fire tests have shown that such behaviors are strongly affected by installation practices (e.g., the method of sealing joints). Furthermore, the physical properties of the barriers needed to address such complex issues are not readily available.

For both inter- and intra-compartment barriers, it appears that a probabilistic model which combines deterministic modeling with empirical evidence (from both field observations and qualitative tests) is needed. A particular issue that may need to be addressed is that of penetration seals; questions have been raised concerning the effectiveness of these seals in preventing fire spread.

3.2.4 Fire Detection and Suppression

Within the context of an FRA, the objective of a detection and suppression analysis is to determine the likelihood that a fire will be detected and suppressed before the fire can damage critical equipment. This requires an assessment of the performance of automatic systems and of the effectiveness of manual fire fighting efforts.

Ref. 33 describes a methodology which assesses the likelihood of various detection/suppression scenarios and their associated suppression times using generic fire protection system reliability estimates and detection/suppression time data obtained from nuclear power plant fire events. The results obtained using this methodology are presented in Ref. 34 and have been used in a few FRAs (e.g., [35]). An alternate methodology which: a) does not explicitly identify different detection and suppression scenarios, b) uses physical models included in FPETOOL [36] to estimate detector and sprinkler actuation times, and c) uses expert judgment to estimate other characteristic delay times in the fire detection/suppression process, has been used in the LaSalle FRA [9].

Most FRAs have used a simpler model in which automatic systems, if they are credited and actuate, are assumed to be immediately effective. (See the guidance provided in Ref. 16.) The results of calculations for equipment damage times are sometimes compared with the results of FIVE worksheet calculations for fire detector and sprinkler actuation times to determine if automatic systems should be credited. If automatic suppression is unsuccessful, the likelihood that manual suppression efforts will be effective before equipment damage is then determined. A possible weakness with this simpler model is its neglect of delays in fire suppression following fixed system actuation observed in real events (e.g., the Browns Ferry fire). However, because the fire growth models used in FRAs do not account for the retarding effects of suppression activities, the risk impact of this neglect is not clear.

Regardless of the methodology employed, detection and suppression analyses require estimates of the reliability of automatic detection and suppression systems. Current FRAs use generic industry (non-

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[2]The analysis considers fire doors, dampers, and penetrations, but does not explicitly define what is meant by "barrier failure."

nuclear as well as nuclear) estimates which can account for plant practices (e.g., installation and maintenance) in only an average manner. For example, in the case of detection systems, the estimates cannot account for such plant- and scenario-specific factors as detector actuation logic, detector location, detector spacing, room congestion, and the behavior of the fire. Similar concerns hold for automatic suppression systems. It is important to note that the suppression system reliability estimates are generally based upon data for system actuation. Because they do not address the issue of suppression system effectiveness, they are not direct measures of the likelihood of successful suppression (prior to damage). It is also important to note that, even if it can be assumed that suppression system actuation is equivalent to fire suppression prior to damage, the use of generic suppression system reliability estimates may be optimistic in studies where severity factors are used in the fire initiation analysis (see Section 3.1). This is because the reliability estimates are not conditioned on the fire severity.

In addition to fire protection system reliability estimates, detailed detection and suppression analyses also require estimates of the delay times (e.g., the detection time, the time to initiate fire suppression, the time to final suppression) characteristic of the fire suppression process. More precisely, since these times should be modeled as random variables, estimates of the parameters of the aleatory distributions for these times are required. As indicated above, currently available methods for estimating these parameters involve the use of empirical event data, simple physical models, or expert judgment.

Regarding event data, two key issues are the availability of data and the applicability of the data to the scenario being analyzed. Objective data for detection times (i.e., the time intervals between fire initiation and detection) are, almost by definition, quite rare. Generally, the first indication of the fire is when the fire is detected either by automatic detectors or by plant personnel. (Occasionally, the fire initiation time can be inferred from detailed event narratives.) Suppression time data are more available, but are not reported for all fire events. The data are generally insufficient to show how the suppression time distribution varies as a function of such issues as the location, severity, and accessibility of the fire. (Note that Ref. 34 presents different distributions for "high" and "low" severity fires, but this categorization depends on a somewhat subjective interpretation of event narratives.)

Regarding model-based approaches for estimating event timing, the same concerns discussed in Section 3.2.1 apply here as well. In particular, the accuracy, limitations, and uncertainties in FRA physical models with respect to predicting smoke and temperature levels for realistic power plant scenarios are unclear. It is important to observe that fire models which are conservative with respect to fire damage predictions may be non-conservative with respect to fire suppression. Furthermore, the use of one fire model in the damage analysis and a different fire model in the suppression analysis can lead to significant errors in the prediction of damage likelihood.

Expert judgment, often supported by the results of plant fire brigade drills, has been used in many FRAs to estimate the time to manual suppression. The analyses typically assume that the manual suppression time equals the brigade arrival time and often do not account for delays associated with detection (prior to brigade activation) or actual fire suppression (following brigade arrival). They also typically do not address aleatory uncertainties associated with the suppression process, e.g., variations in response time due to the time of day. The LaSalle FRA [9] addresses these concerns to some extent by using expert judgment to estimate the minimum, maximum and average times to detection, suppressant application, and suppression (or substantial control) for a variety of scenarios. However, the LaSalle FRA has the same basic problem as other FRAs using expert judgment in the detection and suppression analysis; it does not reflect actual delay times from previous events.

The preceding discussion addresses estimation issues in detection and suppression analysis. Refs. 2, 8, and 19 raise a number of modeling issues which are not quantitatively addressed by most FRAs. These include the impact of smoke and loss of lighting on the effectiveness of manual fire fighting, the effectiveness of compensatory measures (e.g., fire watches) for temporary fire protection deficiencies, and the effect of interactions between the fire growth and suppression processes on the likelihood of suppression before damage. The first issue includes the possibility of misdirected suppression efforts which can damage sensitive plant equipment; as indicated in Section 3.2.2, some but not all of the information needed to address this issue is presented in Ref. 29. The first issue also includes the possibility that scenario-specific smoke and loss of lighting effects will require modifications to the generic suppression time distributions used in many FRAs. The second issue stems from the observation that a number of FRAs assume that fire watches are as reliable as automatic systems in suppressing fires regardless of the fire characteristics. There currently is no technical basis to confirm or refute this assumption. The third issue arises from the fact that current FRAs do not account for the inhibiting effects of suppression activities on fire growth and often do not account for the reduction in fire suppression probability as fire severity increases.

The general modeling framework described in Ref. 33 appears to contain all scenarios addressed by other FRA detection and suppression analyses, and also appears to be capable of incorporating treatments of most of the issues discussed above. (The main exception is the interaction of the fire growth and suppression processes.) The implementation of this framework, however, does not yet address many of these issues. It appears that improvements on the implementation, including the use of information employed by other approaches (e.g., the predictions of physical models for detection and suppression, the results of fire brigade drills), are needed. Note that this framework is not suitable for dealing with detailed fire growth and suppression interactions; if these must be treated (e.g., in non-FRA applications), a more simulation-based approach will be needed.

3.3 Plant System Response Analysis

For each fire scenario involving damage to a set of equipment, the FRA must assess the conditional core damage probability (CCDP). This analysis must address the response of plant hardware and staff under fire conditions. It should be noted that FRAs which use internal events analyses without modification to assess the CCDP do not address many of the issues raised in this section.

Regarding the hardware response, a potential concern is the independence of those systems and components which are not directly affected by the fire. For example, will the fire cause cascading electrical faults which will disable other equipment and safety functions? While many plants have considered this issue deterministically, it is not clear that a system reliability analysis (which allows for failures of components with some probability) would dismiss the importance of such a scenario.[3] This concern, as well as related concerns regarding main control room fires [e.g., the loss of control power before the transfer of control from the main control room to the remote shutdown panel(s)] and spurious actuation of equipment leading to component damage or LOCAs, have been discussed under the general title of "control systems interactions" by Ref. 8 and have been classified as Generic Safety Issue 147 (GSI 147): "Fire-Induced Alternate Shutdown/Control Room Panel Interactions." Reviews of recent IPEEEs indicate that the risk associated with this concern is still not well understood [37].

A second concern is with the likelihood that safe shutdown equipment not directly affected by the fire will actually be available when called upon. Appendix R to 10 CFR 50 requires that "one train of equipment necessary to achieve hot shutdown from either the control room or emergency control station(s)

[3]Note that many of the deterministic circuit analyses have apparently not been done to a sufficient level of detail to assure correct functioning in the event of a fire, even if no random failures are considered [37].

must be maintained free of fire damage by a single fire, including an exposure fire." However, it does not provide any requirements concerning the availability (or, for that matter, the reliability) of this equipment. As shown by the Quad Cities IPEEE [11], situations where the equipment unavailability is significantly higher than the generic values typically used in PRAs can be important contributors to risk.

Regarding the response of plant operations staff to fire events,[4] current FRAs treat the effect of fires in relatively crude ways. Some FRAs increase human error probabilities to account for the additional "stress" induced by the fire and some do not take credit for ex-main control room actions in the affected fire area (due to heat and smoke). However, these adjustments may not adequately address such plant-specific issues as the role of fire brigade members in accident response or the complexity of fire response procedures,[5] nor are they universally agreed upon. Moreover, they are quite judgmental; there currently is no strong technical basis for the magnitude (or even direction) of the adjustments.

Another concern with the treatment of operator response involves "errors of commission." As is true with PRAs in general, FRAs do not address these errors very well. In particular, they do not address possible effects of fire (including fire-induced faulty instrumentation readings and spurious equipment actuations) on operator situation assessment and decision making, nor do they address incorrect operator actions stemming from incorrect decisions. Using the terms of Ref. 38, FRAs do not address the likelihood of "error forcing conditions" being caused by a fire or the likelihood of "human failure events," given these error forcing conditions.

From the standpoint of research needs identification, neither of the hardware concerns appears to require any methods development; some analysis is required to determine their risk significance with respect to the industry, as well as with respect to individual plants. On the other hand, methods development is required to improve the treatment of operator behavior under fire conditions. An empirical basis for adjusting the results of conventional human reliability analyses and a practical approach for assessing the significance of fire-induced errors of commission are required. Research relevant to the latter area is ongoing (e.g., [38,39]); the results of these efforts need to be applied in an FRA context.

3.4 Scenario Risk Assessment Issues

The first six sets of issues listed in Table 3 (Issues I1-I4, E1-E4, H1-H4, B1-B4, S1-S5, and P1-P5) have been identified largely through an examination of the current FRA paradigm [as represented by Eq. (1)]. The next set of issues listed in Table 3 (Issues R1-R12) have been identified through a variety of other means, including reviews of FRA treatments of specific scenarios, the results of previous investigations of fire risk assessment issues (e.g., [8]), and input from NRC staff concerning scenarios not currently addressed by FRAs. Most of these issues are associated with integrated assessments of risk for particular scenarios. They are briefly discussed in this section.

Main control room fires. Main control room (MCR) fires have been shown to be dominant contributors to risk in some FRAs and negligible contributors in others. Unfortunately, much of this difference in predicted risk significance appears to be due to modeling assumptions about the likelihood of severe fires in the MCR,

[4]Note that human performance issues concerning fire detection and suppression are discussed in Section 3.2.4.

[5]Work on self-induced station blackout (SISBO) and a number of recent IPEEE studies appear to indicate that complexities in procedures designed to mitigate possible fire-induced hot shorts can be significant contributors to risk.

the time available to suppress a severe fire before MCR evacuation is required, and the likelihood of successful operator actions given a severe fire. There currently is insufficient information available to specify how MCR fires should be modeled; improved methods and data are needed to reduce the degree of analyst-to-analyst variation in the results.

Turbine building fires. Historical turbine building fires (e.g., the Narora fire [21]) and the Quad Cities IPEEE [11] show that severe turbine building fires can be important contributors to risk. Potential concerns with the adequacy of FRA tools for these fires have been mentioned earlier. They include the lack of knowledge concerning the frequency-magnitude relationship for turbine building fires (see Section 3.1) and the adequacy of current FRA tools for predicting the environment induced by a severe turbine building fire (see Section 3.2.1). Partly because of these concerns, the overall risk contribution from turbine building fires at any given plant is uncertain.

Containment fires. The containment contains safety-related equipment (e.g., cables for redundant instrumentation) which might be vulnerable to a severe fire. However, most FRAs have assumed that containment fires are negligible contributors to risk (even for non-inerted containments) per the arguments stated in Ref. 13, i.e., containment fires are infrequent and previous FRAs have shown that containment fires are not risk significant. Noting that most previous FRAs have not explicitly addressed fire-induced instrumentation failures and many have not addressed spurious equipment operation, the latter argument may be questionable. An improved assessment of the potential risk contribution of containment fires is needed. If a detailed analysis is required, improvements in the state-of-knowledge concerning the frequency-magnitude relationship for containment fires and improved tools for predicting fire environment within containments will be needed.

Seismic/fire interactions. Ref. 8 identifies a number of issues associated with the effect of seismic events on fire protection and fire risk. These include seismically-induced fires (e.g., fires involving the tipping of improperly anchored electrical cabinets) and seismically-induced suppression system actuations. A recent investigation of the effects of the January 17, 1994 Northridge earthquake on industrial facilities (including conventional power plants) appears to indicate that suppression system actuations are more likely than fires [40]. (Fires only appear to be a significant concern when the earthquake causes the failure of flammable gas lines.) Note that according to Ref. 40, the peak ground accelerations associated with the Northridge earthquake were much larger than the design values of many of the facilities examined. Ref. 8 indicates that the risk associated with seismic/fire interactions can be addressed via dedicated walkdowns; however, it does not provide a methodology for quantifying the risk associated with walkdown findings.

Multiple units. The results of a number of FRAs have shown that some multi-unit sites have areas where a single severe fire can initiate transients and damage mitigating equipment for multiple units. Another, more subtle multi-unit interaction involves situations where safe shutdown of one unit requires equipment from another unit. Besides depriving the "non-affected unit" of the services of that equipment, errors in performing the actions required to make the equipment available to the "affected unit" could lead to further unavailabilities of the non-affected unit's equipment. It appears that most (if not all) FRAs to date have focused on the fire risk associated with a single unit; the frequency of multiple unit core damage due to a single fire has not generally been explicitly calculated. The detailed results of the Quad Cities IPEEE [11] indicate that, at least for some plants, this frequency may not be negligible. The current FRA framework is capable of dealing with this issue. However, detailed examinations of the overall plant response and modifications in the plant response analysis models are needed to assess its risk significance.

Non-power and degraded conditions. Most current FRAs have focused on the fire risk associated with at-power operation. The fire risk associated with low power and shutdown operation has received limited attention (e.g., [35]). The fire risk associated with scenarios involving: a) damage to equipment required to achieve and maintain cold shutdown, or b) degraded conditions (i.e., fires following a non-fire initiating event) has apparently not been addressed. The issue of degraded conditions is potentially a concern for consequential fires, e.g., fires caused by the same chain of events which leads to a loss of offsite power. The current FRA framework appears to be capable of dealing with non-power and degraded conditions. Analyses which reflect possible changes in fire frequencies (and in the frequencies of severe fires), as well as changes in plant response, may need to be performed. Note that Ref. 12 presents information useful for the quantification of fire frequencies during low power and shutdown operation.

Decommissioning and decontamination. FRAs have not been performed to assess the risk associated with the decommissioning and decontamination phases of a plant's life cycle. If fire-induced direct releases of radioactive material to the environment or occupational risks need to be analyzed, additional FRA methods and data may be needed.

Fire-induced non-reactor radiological releases. As shown by Eq. (1), current FRAs are focused on evaluating scenarios involving core damage. The risk associated with direct radiological releases to the environment has not yet been evaluated. Note also that the impact on core damage frequency due to direct radiological releases (which can affect operator performance) is not evaluated in current FRAs.

Flammable gas lines. Potential problems with the leakage and ignition of combustible gases within plant compartments are addressed under Generic Safety Issue 106: "Piping and the Use of Highly Combustible Gases in Vital Areas." As analyzed in Ref. 41, this is a medium priority generic issue. Based upon the IPEEE reviews to date [37], it is not known if this issue is highly risk significant for any single plant.

Scenario Dynamics. As pointed out in Ref. 42, the timing of fire-induced equipment failures (which can be on the order of tens of minutes for some scenarios) is not treated in current FRAs. Instead, the FRAs treat fire-induced equipment failures as occurring at the beginning of the scenario. Furthermore, they effectively assume that the operators know exactly what has been lost due to the fire. In an actual fire, of course, equipment can be lost progressively over the course of the scenario, and the operators will not necessarily know exactly what has been lost (or what indications to mistrust) at any point in time, let alone what will be lost in the future. The current FRA approach can be conservative in situations where the equipment is lost well after it is truly needed. It can be non-conservative in situations where the scenario dynamics introduce considerable confusion. In general, the scenario dynamics could present a very different context to the operator than the one assumed in FRAs. The effect of this different context on operator performance and predicted risk could be significant [38,39].

Precursor analysis methods. The NRC's accident sequence precursor program, which evaluates the risk significance of reported events and plant conditions as precursors to core damage accidents, currently lacks tools for evaluating fire events or conditions involving fire protection deficiencies. Tools for performing such evaluations have been proposed (e.g., [43]) but not yet rigorously tested.

Uncertainty analysis. A meaningful uncertainty analysis requires a careful consideration of uncertainties in models, as well as in model parameters. The issue of model uncertainty is discussed in Section 3.2.1. It is worth noting that a proper treatment of uncertainties can significantly affect perceptions concerning the credibility of current FRAs. Ref. 19 uses the results of a formal uncertainty analysis to show that, from the perspective of FRA, the need for extremely accurate fire growth models may be significantly less than implied by the results of sensitivity calculations of the kind discussed in Ref. 8.

3.5 Other Issues

The last four issues listed in Table 3 (Issues O1-O4) concern general means to improve FRA and fire risk management. The first two involve the need to collect information from past events and from other fire research efforts. Regarding past events (Issue O1), serious fires have occurred in U.S. and international nuclear power plants, as well as in other industrial facilities. Current FRAs tend to make limited use of the information obtained from these events. For example, they use counts of events to estimate fire frequencies, but do not use event descriptions to determine if changes in the basic FRA structure are warranted. Regarding other fire research efforts (Issue O2), a substantial amount of work is being conducted outside the nuclear industry. For example, Ref. 44 reports on an international effort to validate current fire simulation software. The results of these validation efforts, or other non-nuclear fire modeling activities (e.g., [24,45]) have not yet been generally reflected in current FRAs. While issues O1 and O2 do not imply specific research needs, they indicate elements that need to be incorporated in a viable fire research program.

The second two issues in Table 3 concern the use of FRAs in risk-informed, performance-based regulation. Issue O3, "Comparison of methodologies," refers to the fact that a number of different methodologies are used by current FRAs. The degree to which the differences in FRA results are due to these methodological differences (which affect analysis level of detail, modeling assumptions, and data) is unclear. Clearly, this source of variability needs to be better understood when the FRAs are used to support regulatory decision making. Issue O4, "Standardization of methods," is a natural follow-on to Issue O3. It concerns the degree to which FRA methods and data can be or should be standardized.

4. CONCLUDING REMARKS

Improvements in the NRC staff's ability to thoroughly understand and accurately evaluate nuclear power plant fire risk require efforts in a number of areas. In order to initiate improvements in these areas, this paper has developed and discussed a list of potential research issues which involve: research on material properties and scenario phenomenology, the development of methods and tools based on the results of this research, and the application of these methods and tools to actual plants. The next steps in the improvement process are the development of a prioritized list of research topics (where one topic may include a number of related research issues) and the development of a research program to address these topics. Work on these steps is ongoing.

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PROBABILISTIC SAFETY ANALYSIS OF NUCLEAR MATERIALS

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Abstract

Two programs to develop techniques for the probabilistic safety analysis of nuclear materials are in progress. The first program will investigate the potential public risk associated with the storage of spent reactor fuel in dry casks. The second program is investigating the likelihood and consequences of breaching sealed radioactive sources used in industrial thickness and density gauges.

As the NRC moves toward a risk-informed approach to regulation, probabilistic risk analysis tools are being put to use across the entire spectrum of the Agency's mission. Although most PRAs are studies of power reactor safety, the same techniques can potentially be applied to nuclear materials safety and associated issues. Two programs to develop such techniques currently exist within the Office of Nuclear Regulatory Research: dry cask storage of spent fuel, and sealed industrial sources.

Dry Cask Storage

Dry cask storage is an alternative storage method for spent reactor fuel. After several years of storage in a spent fuel pool, the decay heat from a spent fuel assembly is sufficiently low that the assembly can be cooled by natural convection in air. However, the inventory of radioisotopes is still significant and the radiation field surrounding the assembly still quite hazardous. Dry casks are designed to hold such a fuel assembly, provide adequate shielding, and provide adequate cooling by means of natural convection. The casks are sufficiently robust to be stored on-site but out of doors, and thus are an attractive possibility for relieving the overcrowded state of spent fuel pools, at least in the short term. Nevertheless, there is inevitably some public risk associated with such a large inventory of radioactive material, and it is intended to evaluate this risk in a quantitative fashion. This evaluation is just now beginning.

Because of the relatively low decay heat generation rate and the passive nature of the cooling of the stored fuel assemblies, there are relatively few credible scenarios which lead to melting of the fuel. However, the possibility of the release of activity from mechanical damage and cladding breach must be considered. In addition, an event which leads to severe mechanical damage and pulverizing of the fuel must be considered.

Thus, the first step of the analysis will be to define damage states which can lead to a non-trivial release of radioactive material. This will be followed by a systematic identification of potential accident scenarios, and the quantification of these scenarios. Because there are likely to be few internally-generated accident sequences once the casks are in place for indefinite storage, the analysis will include sequences associated with the loading and transfer of the casks, as well as externally-generated sequences such as those associated with floods, earthquakes, etc.

Sealed Sources

The devices of interest in the current study are sealed radioactive sources used in industrial thickness gauges, density gauges, and similar gauges utilized in industrial process control. Such devices are extremely robust and are very safe even in a severe industrial environment over a period of many years. Because of this, such devices are "generally-licensed." (Generally-licensed devices are devices containing radioactive material, but which are so robust and well-shielded that the manufacturer rather than the user is licensed by the Agency.)

The majority of companies who use sealed sources make conscientious attempts to track their inventory of such devices. Also, the devices are usually painted a distinctive color, and are labeled with the purple trefoil and appropriate warnings. However, there is a vast number of these devices in use. Moreover, in an industrial process environment, almost every piece of equipment turns a uniform brown or grey color after a number of years, regardless of initial paint color or the presence of stickers. Not surprisingly, sources are occasionally lost in spite of a company's best efforts. These lost sources are still not immediately hazardous unless the casing is breached in some way.

A situation which often leads to the loss and breaching of a sealed source occurs when a company goes into bankruptcy. Once the plant is padlocked and the personnel become occupied with finding new employment, administrative controls tend to break down. Later, the plant equipment, which may by this time have experienced a considerable period of neglect, is often sold to a scrap dealer, who simply removes anything of obvious value, cuts the remaining equipment down to convenient sizes, and ships it off to a steel mill.

To protect themselves from radioactive material in the incoming scrap loads, steel mills will generally pass incoming loads (in trucks or rail cars) between large plastic scintillation detectors. The effectiveness of these detectors is not perfect, since the sealed sources are designed to be well shielded, and may well be located in the middle of a load of steel. Moreover, the usual practice when an incoming load trips the monitor alarm is to simply reject the entire load. The scrap dealer may well simply take the same load to another mill and try again, until someone accepts the load.

If a sealed source is melted in a blast furnace, the result is, of course, ingots of contaminated steel. In addition, radioactive material will be taken up in the furnace dust and be collected in the baghouse. What is not taken up by the baghouse will be released into the atmosphere. The furnace dust contains a variety of heavy metals, and is often sold to still other recycling facilities for recovery of valuable material. Thus, if not detected, the contamination can range well beyond the original mill. Even if the contamination is detected at the original furnace, as it usually is, the necessary shutdown and decontamination of the facility may well be so expensive that the mill operator is driven out of business. Not surprisingly, there is some sentiment even within the steel industry for more regulation of such devices, and the current study, if successful, will provide the basis for risk-informed regulatory decisionmaking. The study is now well underway, but will not be complete until late in 1998.

The probabilistic analysis of this is rather interesting in that there are, at least in principle, actual statistics available for the initiating event. (Actual collection of data is not so straightforward, since such incidents are not widely reported, particularly if a load of scrap trips the monitor alarms and the load is rejected. It is planned to collect this data by actually performing a survey.

The effectiveness of the various monitoring stations has never been calculated in any rigorous fashion, since it depends on the location of the source within the load of scrap, the density of the scrap, the nature and shielding of the source, the shape and speed of the transporting vehicle, and the setup of the detectors. It is intended to perform some scoping shielding calculations to estimate the effectiveness of the monitoring stations.

Calculating the radiological consequences is also complicated, because of the rather complex waste stream. Thus, a mathematical model is being built to cover the various waste streams (furnace dust, etc.) and estimate the extent of contamination.

As can be seen, this particular program still has much to do. Hopefully, we will be able to provide updated information at our next meeting.

Development of Data Base with Mechanical Properties of Un- and Preirradiated VVER Cladding

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Analysis of recent RIA test with PWR- and VVER high burnup fuel, performed at CABRI, NSRR, IGR reactors has shown that the data base with mechanical properties of the preirradiated cladding is necessary to interpret the obtained results. During 1997 the corresponding cycle of investigations for VVER clad material was performed by specialists of NSI RRC KI and RIAR in cooperation with NRC (USA), IPSN (France) in two directions:

- measurements of mechanical properties of Zr-1%Nb preirradiated cladding versus temperature and strain rate;
- measurements of failure parameters for gas pressurized cladding tubes.

Preliminary results of these investigations are presented in this paper. Measurements of mechanical properties were done during tensile tests of ring samples manufactured from the cladding of commercial VVER fuel rod irradiated at the 5th unit of NovoVoronezh NPP up to 48 MWd/kg U. Validation of the procedure was done due to special comparative tests with unirradiated Zr-1%Nb cladding independently performed at RIAR (Russia), ANL (USA) and IPSN-CEA (France). Obtained results for the ultimate strength are presented in Fig. 1. After methodological aspects of measurements were completely agreed upon the main stage of tests started. During this stage mechanical properties of preirradiated VVER cladding versus temperature and strain rate were determined. Simultaneously with these tests we have performed the reassessment of the data base with mechanical properties of unirradiated Zr-1%Nb cladding presented in [1]. Dr. Kobyliansky from RIAR prepared the necessary input data for this work. Both data base for un- and preirradiated samples were statistically processed with methods of non-linear regression analysis; and the corresponding regression curves are presented in Figs. 2-4. The purpose of such procedures was to develop analytical models of mechanical properties of Zr-1%Nb cladding that could be used for the computer simulations. The first versions of alternative MATPRO modules describing plastic deformation of VVER cladding were developed.

Analysis of obtained results for both states (un- and preirradiated) of material demonstrates the increasing of strength parameters and reducing of ductility for preirradiated cladding at low temperatures. Effect of strength increase is completely eliminated at the temperatures higher than 860 K, which could be explained by annealing of irradiation damages. As for characteristics of ductility for low strain rates at high temperatures, the most probable explanation is that superplasticity has occurred in both states. At the same time it's evident that additional measurements should be performed in this area in order to finally specify corresponding characteristics.

On the whole obtained results show that VVER preirradiated cladding has high enough ductility margin at low temperatures. This conclusion is illustrated by typical view of the ring samples rupture presented in Fig. 5.Comparison of mechanical properties of Zr-1%Nb and unirradiated Zry-4 [2, 3] presented in Fig. 6 leads to the similar conclusion.

Experimental results on strain rate influence on mechanical properties in range of 0.002 - 0.5 1/s are presented in Fig. 7 in comparison with MATPRO model [2]. In the temperature range above 800 K strong influence of strain rate for both materials was observed. It should be noted a good quantitative agreement of strain sensitivity exponent for both alloys in temperature range 293 - 1200 K. For temperatures above 1200 K additional measurements should be performed.

Other direction of research was focused on obtaining the data base characterizing failure parameters of pressurized VVER cladding tubes or in other words - to obtain the data base characterizing of cladding failure due to ballooning. This research includes the tests of un- and preirradiated VVER cladding tubes. Tests of unirradiated cladding are over by now and are described in this paper. Burst tests of preirradiated tubes will be completed to the end of this year.

Special facility was used for these investigations. The scheme of tests was as following:

- sample of Zr-1%Nb tube 150 mm long was located inside the facility;
- inert gas was supplied to inner plenum of the sample so that the inner pressure was 0.1- 0.4 MPa during the heating up process till the set temperatures (1073 - 1473 K);
- increase of the sample temperature up to the set value was done with electric heater of the facility;
- gas pressure inside the sample was increased up to the burst of tube sample.

Typical post-test view of one of the samples is presented in Fig. 8.

In the frame of this work, parameters of the cladding burst were studied versus both temperature and rate of gas pressure increase inside the sample. Results of the test are presented in Fig. 9, Fig. 10.

Obtained data show that the rate of pressure increase inside the tube sample can significantly influence the parameters of cladding burst, which isn't only the correlation between the temperature and burst pressure but also as it was found out the cladding shape and hoop strain.

These issues will be considered in our next publications in more details. Besides comparison of obtained results with the previously published data was performed for Zr-1%Nb [4] and Zircaloys [2]. Summarized data base presented in Fig. 11. shows that the burst parameters for Zry- and Zr-1%Nb cladding are very similar in high temperature region. There could be some difference in the temperature level at 1173 K because Zr-1%Nb alloy has completed $\alpha \to \beta$ phase transformation in this area.

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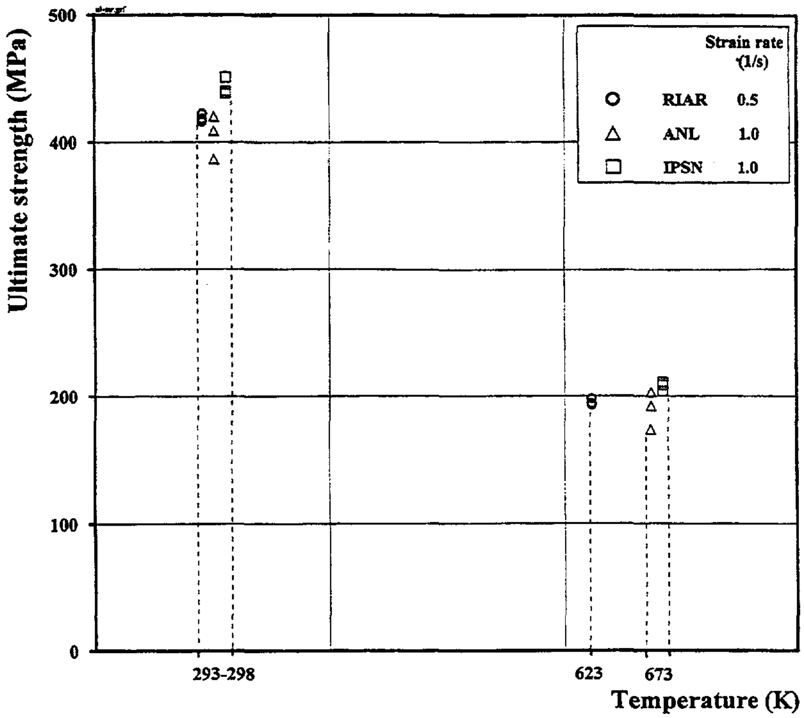


Fig.1 Comparative measurements of ultimate strength

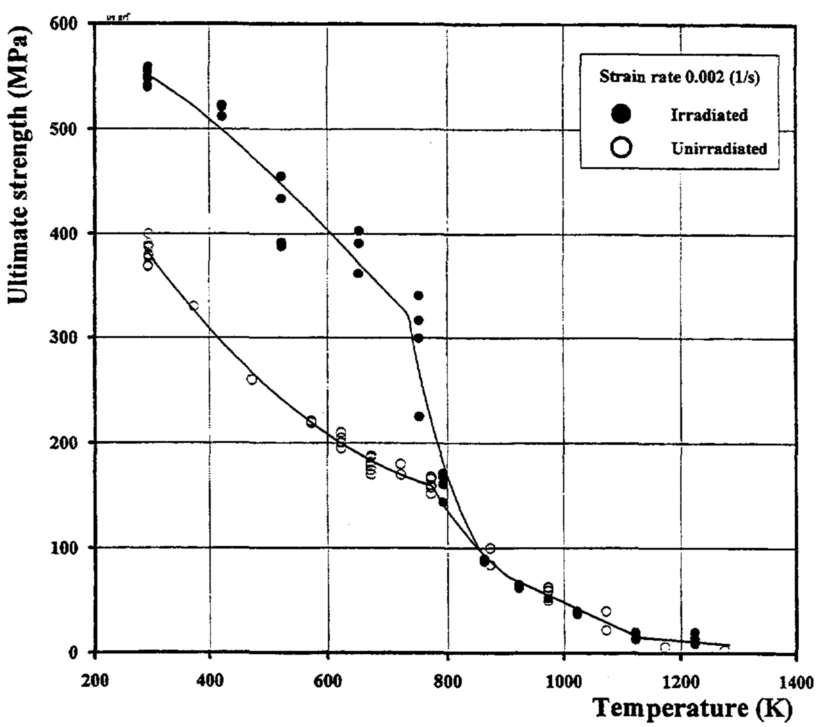


Fig.2 Ultimate strength of Zr-1%Nb cladding vs temperature and fuel burnup (0; 48 MWd/kgU)

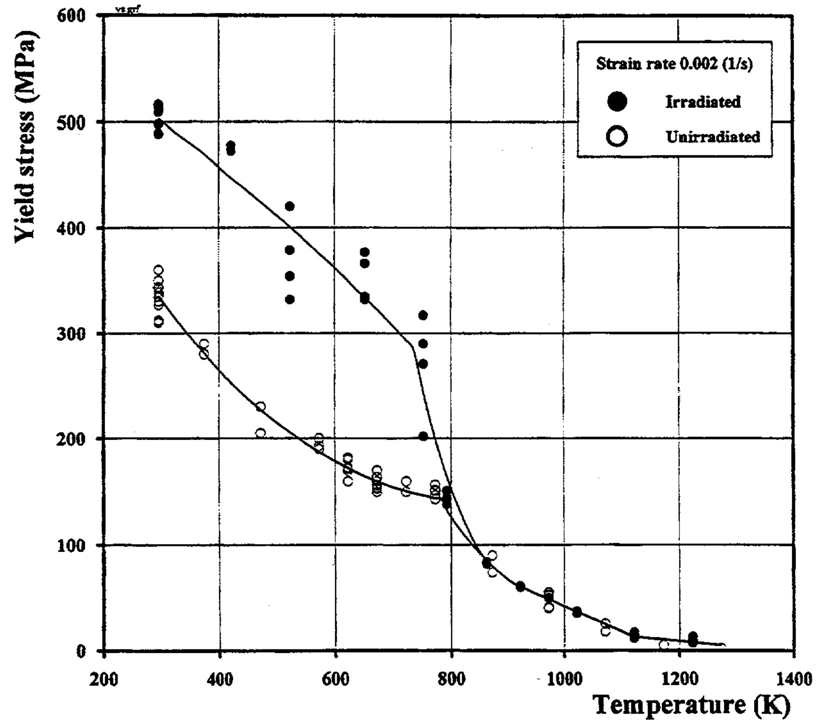


Fig.3 Yield stress of Zr-1%Nb cladding vs temperature and fuel burnup (0; 48 MWd/kgU)

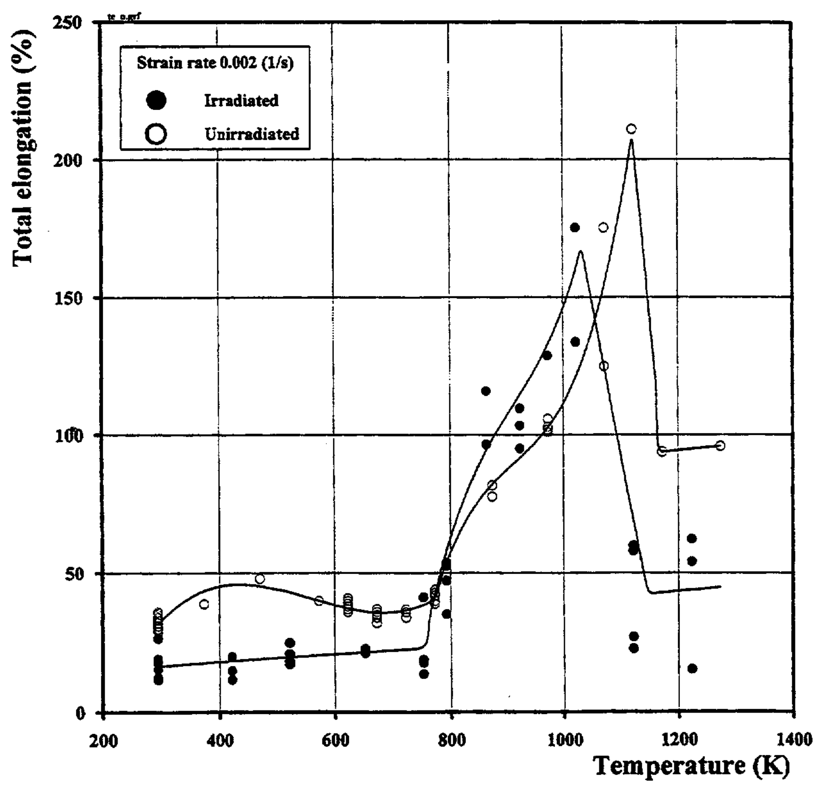


Fig.4 Total elongation of Zr-1%Nb cladding vs temperature and fuel burnup (0; 48 MWd/kgU)

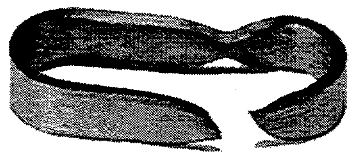
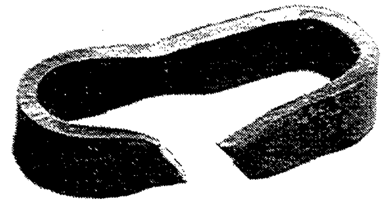
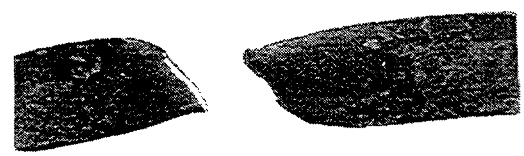
Unirradiated T = 293 K	Irradiated T = 293 K
	
	

Fig.5 Appearance of ring samples after tests

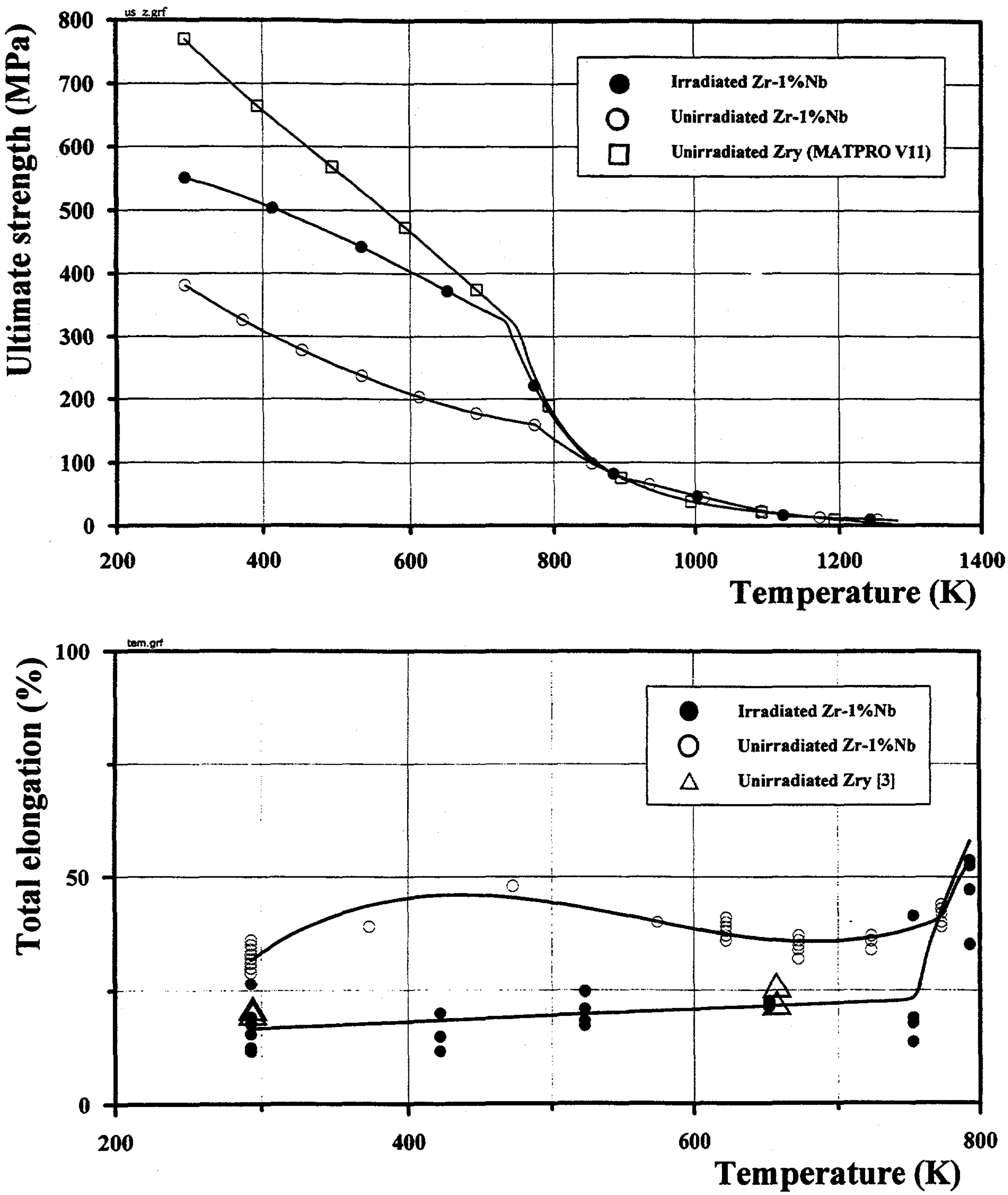


Fig.6 Comparison of Zr-1%Nb and Zry mechanical properties

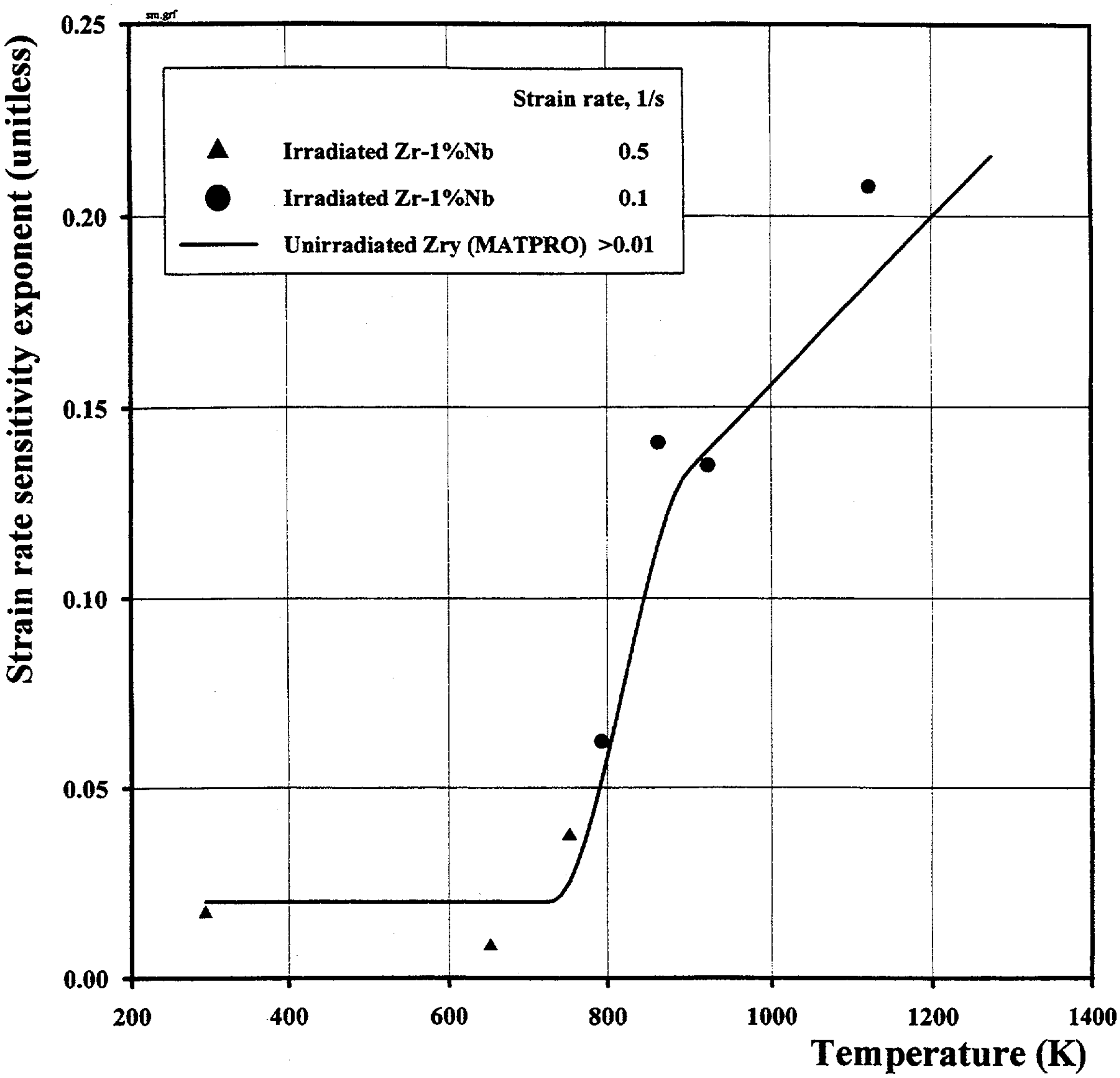


Fig7. Comparison of Zr-1%Nb and Zry strain r

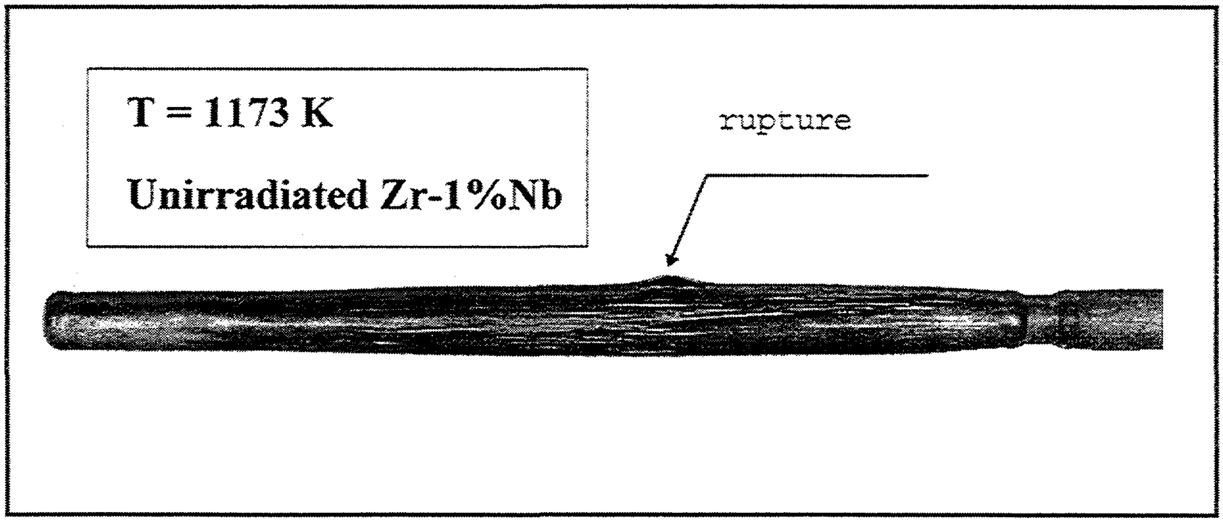


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Fig.8 Appearance of gas pressurized sample after test

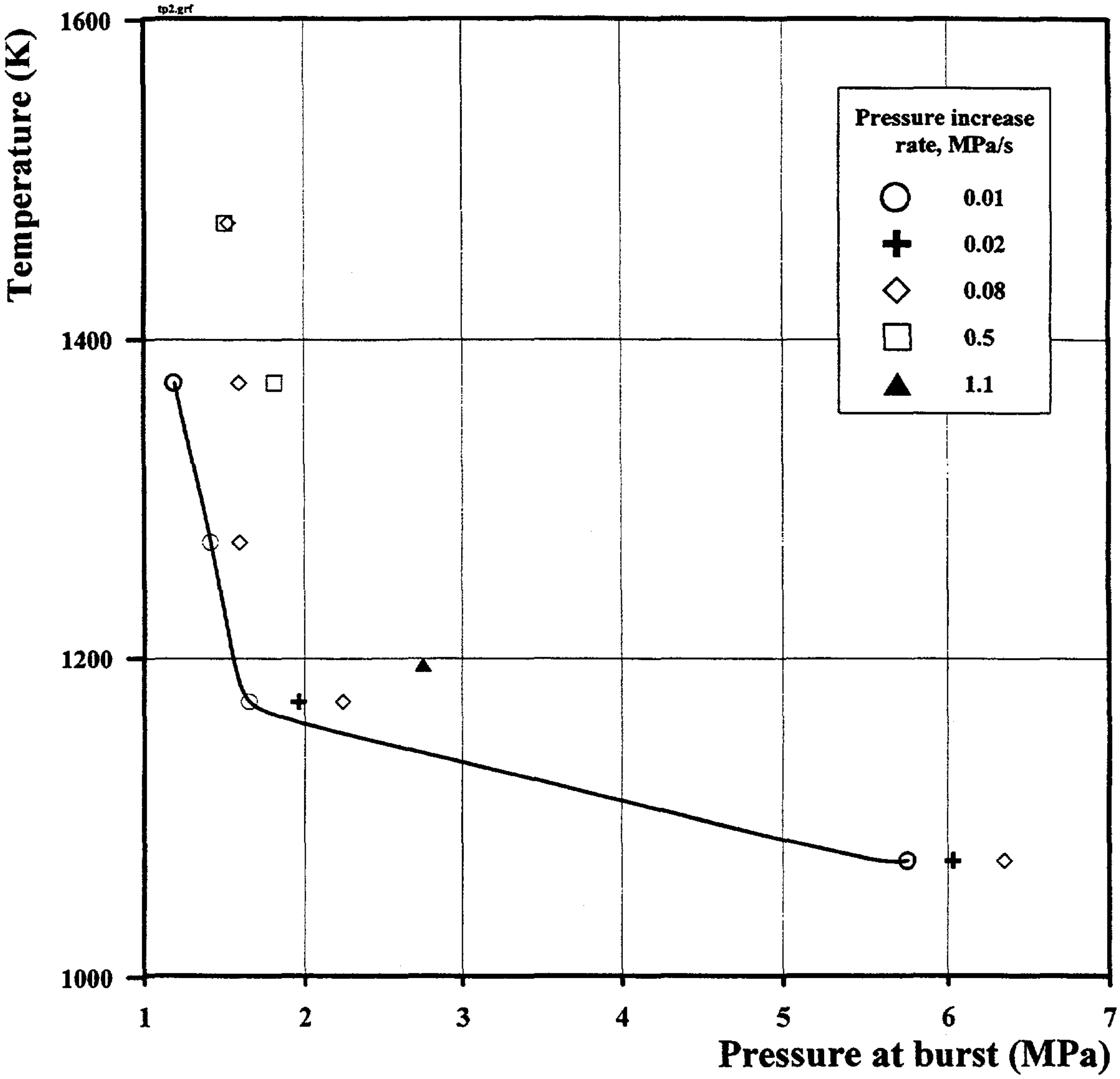

Fig. 9 Relation between burst pressure and temperature for unirradiated Zr-1%Nb

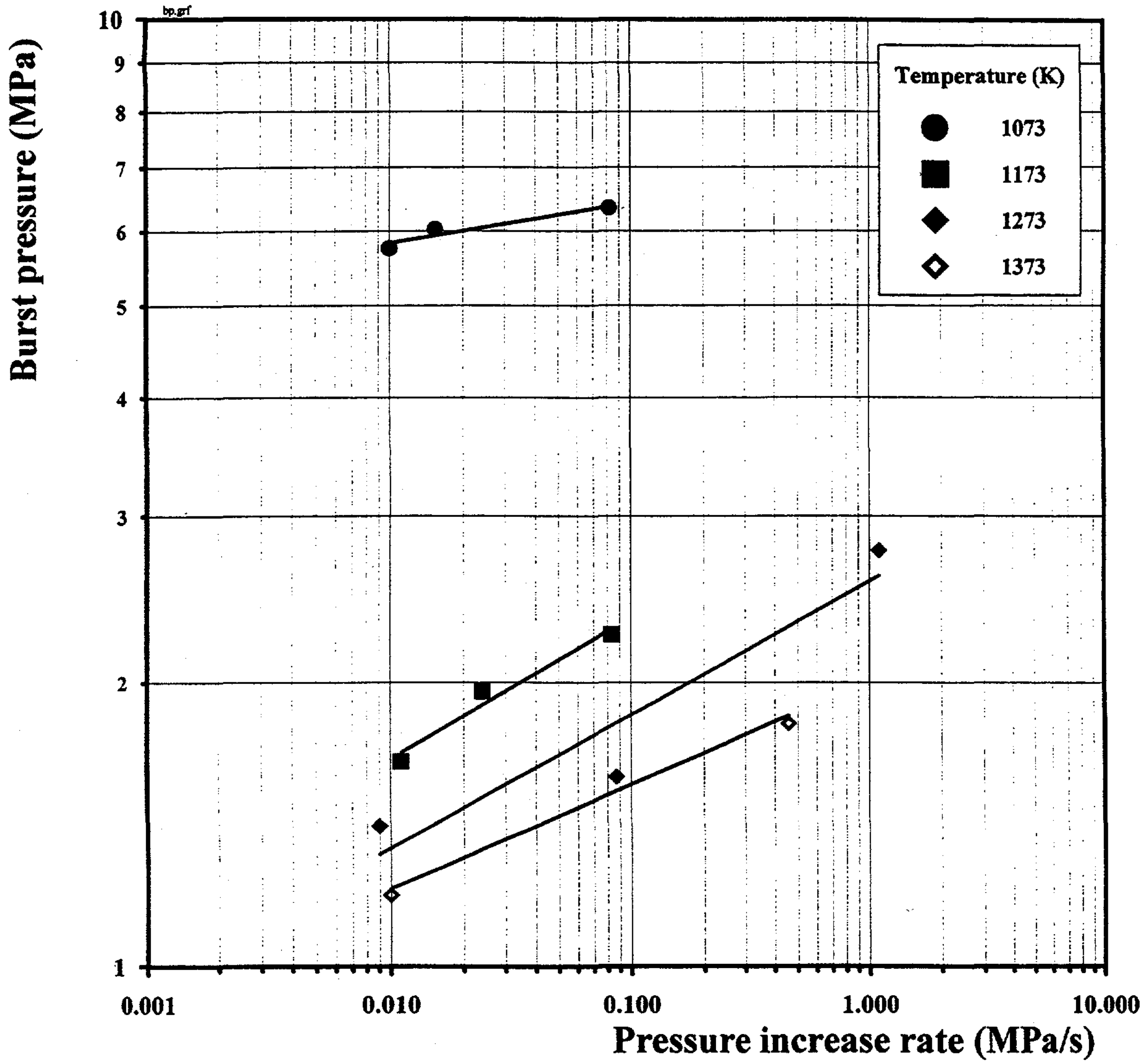

Fig.10 Burst pressure vs pressure increase rate and temperature for unirradiated Zr-1%Nb tube

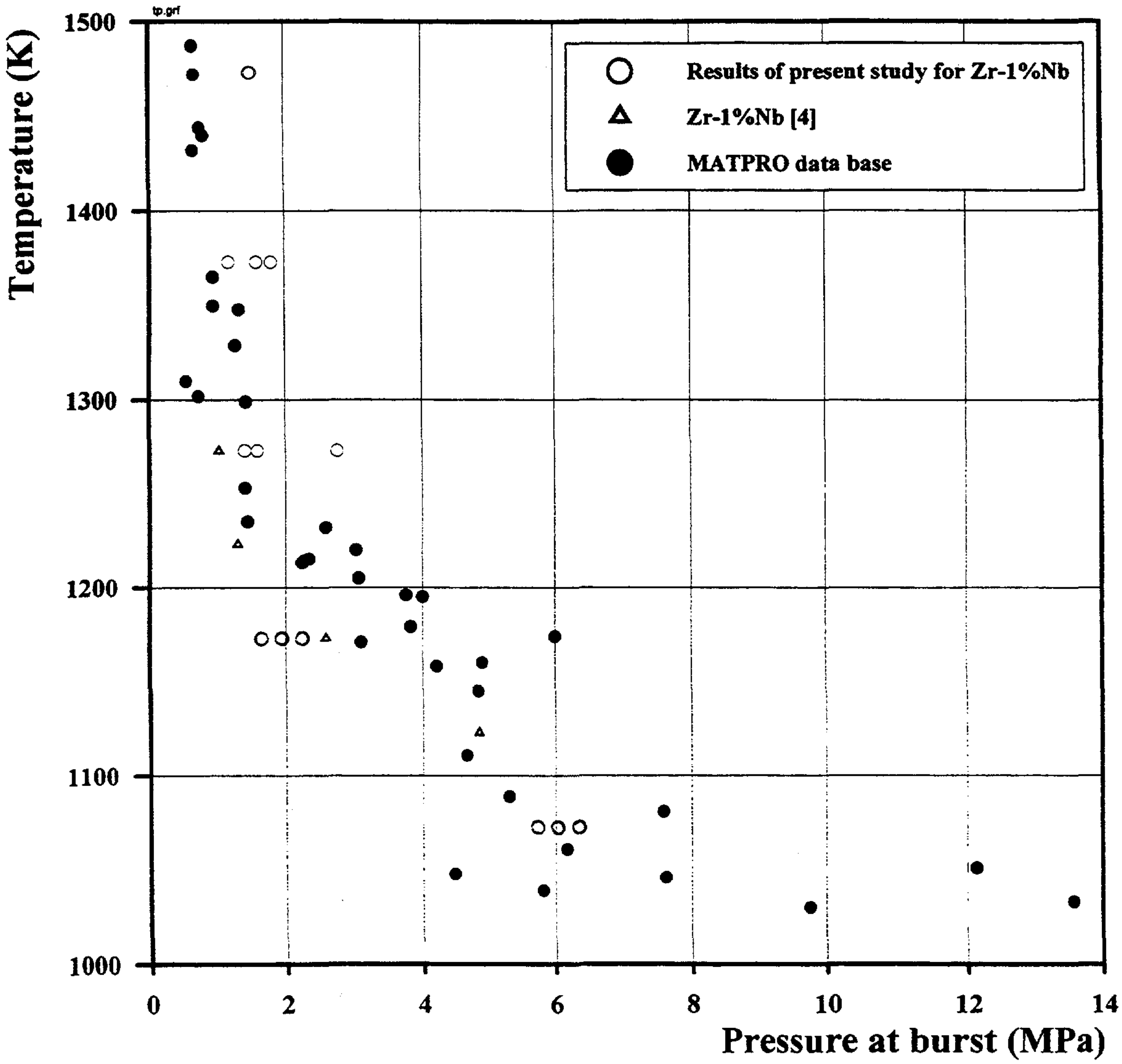

Fig.11 Data base with test results of gas pressurized tube. Pressure and temperature at burst

Modified Ring Stretch Tensile Testing of Zr-1Nb Cladding

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Abstract

In a round robin effort between the U. S. Nuclear Regulatory Commission , Institut de Protection et de Surete Nucleaire in France, and the Russian Research Centre-Kurchatov Institute, Argonne National Laboratory conducted 16 modified ring stretch tensile tests on unirradiated samples of Zr-1Nb cladding, which is used in Russian VVER reactors. Tests were conducted at two temperatures (25 and 400°C) and two strain rates (0.001 and 1 s^{-1}). At 25°C and 0.001 s^{-1}, the yield strength (YS), ultimate tensile strength (UTS), uniform elongation (UE), and total elongation (TE) were 201 MPa, 331 MPa, 18.2%, and 57.6%, respectively. At 400°C and 0.001 s^{-1}, the YS, UTS, UE, and TE were 109 MPa, 185 MPa, 15.4%, and 67.7%, respectively. Finally, at 400°C and 1 s^{-1}, the YS, UTS, UE, and TE were 134 MPa, 189 MPa, 18.9%, and 53.4%, respectively. The high strain rate tests at room temperature were not successful. Test results proved to be very sensitive to the amount of lubrication used on the inserts; because of the large contact area between the inserts and specimen, too little lubrication leads to significantly higher strengths and lower elongations being reported. It is also important to note that only 70 to 80% of the elongation takes place in the gauge section, depending on specimen geometry. The appropriate percentage can be estimated from a simple model or can be calculated from finite-element analysis.

Introduction

As part of a round robin effort between the U. S. Nuclear Regulatory Commission (NRC), Institut de Protection et de Surete Nucleaire (IPSN) in France, and the Russian Research Centre-Kurchatov Institute, Argonne National Laboratory (ANL) conducted 16 ring stretch tensile tests on unirradiated samples of Zr-1Nb cladding used in Russian VVER reactors. We used the "modified" ring stretch test discussed by Arsene and Bai; slightly different geometries were used by the Russian and French researchers. The advantages of the modified ring stretch tests are that the specimens have a well-defined gauge length and measures are taken to minimize the bending moment in the gauge section during stretching of the ring.

The tests were conducted to determine the circumferential (or hoop) tensile properties of unirradiated fuel cladding, and the results will be used to develop procedures for conducting similar tests on irradiated cladding segments in an NRC program for determining the properties of high-burnup LWR cladding under LOCA and other transient-related conditions. This paper discusses the results and analysis of the tests conducted by ANL.

Experimental

In the modified ring stretch test, three inserts are placed inside a ring cut from a cladding tube: two inserts on which the tensile pulling force is applied, and a dumbbell-shaped central spacer to minimize specimen bending during the test. A schematic diagram of the specimen configuration is shown in Fig. 1a, and the fixtures used for attaching the inserts to the tensile machine are shown in Figs. 1b and 1c. The clearance between the three inserts and the specimen is very small (<0.025 mm). All components were made of 17-4 PH stainless steel, which was hardened at 482°C for 1 h in argon. The surfaces of the inserts and spacer were coated with Molykote Z, a dry molybdenum disulfide powder, to minimize friction between the components and the specimen. As will be discussed in the Results section, the presence of friction can have a significant effect on the measured mechanical properties.

The ring specimens were machined by electro-discharge machining (EDM) to provide a narrowed gauge section in the circumferential direction; a schematic diagram of one gauge section is shown in Fig. 2. The figure identifies, by letter, the specimen dimensions: G and L are actually arc determinations; the chord is measured from a projection of the specimen and the radius is used to calculate the arc length. The arc length is used for calculations of strain.

Two specimen designs were used. The first ("A") was derived from relations and diagrams in a report by Josefsson and Grigoriev, who used the modified ring stretch test to study the mechanical properties of irradiated cladding at Studsvik. The second ("B") was a result of a redesign effort because of some questions related to the results generated from the first design. The original design was not optimal because the wider gauge section led to an inhomogeneous plastic strain distribution, and it was thought that such a distribution was leading to higher than expected strains. Finite-element analysis was used to optimize the uniformity of strain distribution in the gauge section while still allowing for the use of the same fixtures and insert components for a second series of tests. The dimensions for each design, referenced to the labels in Fig. 2, are given in Table 1.

Table 1. Specimen Dimensions for VVER Tests (in mm)

Dimensions	Design A	Design B
W	5.03	4.27
w	2.03	1.7
L	4.27	4.27
G	1.7	2.11
r	1.28	1.08

The aspect ratio of the original design was less than 1, while the redesign was approximately the inverse of design A.

The tests were conducted on a servohydraulic tensile machine (Instron Model 1125) in an air atmosphere at two temperatures (25 and 400°C) and two strain rates (0.001 and 1 s^{-1}). The elevated-temperature tests were conducted in a resistance heating furnace, which took approximately 1 h to reach equilibrium at the test temperature. The load as a function of time was recorded on a strip chart recorder. For the low-strain-rate tests, the load signal was also captured through an analog-to-digital converter on a IBM PC computer file, which allowed for subsequent analysis of the data and derivation of the stress-strain curves for each test. For the high-strain-rate tests, because of the short duration of the test and the high frequency required for data capture, the load as a function of time was recorded on a high-speed oscilloscope (Lecroy Model 9354 TM Wavedesc), rather than on the IBM PC. The oscilloscope file was then converted into a file readable by a spreadsheet.

Analysis

The mechanical property results from the tests were determined from the load vs. time curve documented on the strip chart. Ultimate tensile strength was calculated from the maximum load on the strip chart divided by the nominal cross-sectional areas of the two gauge sections (see Fig. 2). The yield strength was determined by drawing a line parallel to the elastic portion of the curve but offset by an amount equivalent to 0.2% plastic strain. Uniform elongation was determined by drawing a line parallel to the elastic portion of the curve but intersecting the curve at the maximum load, then converting the elapsed time to a distance by knowing the chart speed. A similar procedure was used to determine the total elongation, except that the parallel line intercepted the curve at the breaking point of the specimen.

Because the specimen dimensions in both designs were nonstandard, we could not assume that all elongation during a test took place in the gauge sections. Therefore, it was necessary to estimate the percentage of strain that occurred in just the gauge sections during the test. A very close value can be estimated by using finite-element analysis; however, because the standard deviation of the strain values for some of the early tests was large, we felt that an approximate measure would be satisfactory. An approximate value for the percentage of strain that occurs in just the gauge sections was determined by using a simple iterative model as described in the following. We assumed that a certain percentage (say, 50%) of the elongation occurred in the gauge sections. We divided the shoulder (or curved) region into five strips of equal height (measured circumferentially) and calculated the stress in each segment for the maximum load. From the stress/strain curve, we then determined the strains for each stress. The calculated elongations in each segment of the shoulder were subtracted from the uniform elongation read from the strip chart. We assumed that no strain occurred outside of the shoulder and gauge sections; therefore, the remaining elongation must have been in the gauge sections. This resulting percentage of elongation was then compared to the original assumed percentage.

The process was repeated with the assumed percentage set equal to the resulting percentage from the previous iteration until the assumed and resulting percentages in a given iteration differed by no more than 0.1. By using this iterative process, we determined the percentage of strain that occurred in the gauge sections for each test specimen. To determine the uniform strain, the percentage calculated by the iterative model was multiplied by the uniform elongation (up to the point when necking occurs), and the uniform strain was calculated as the percentage increase to the original gauge length that resulted in the new uniform elongation.

The total elongation measured on the strip chart was then used to calculate the total strain, or percent total elongation. The ratio of total elongation to uniform elongation was assumed to be the same as the ratio of total strain to uniform strain, and the total elongation was assumed to occur in both gauge sections at the same time.

Results

Sixteen modified ring stretch tests were conducted; the test number corresponds to the order in which the test was run. Post-test analysis of each test followed the paradigm given in the Analysis section. From the load vs. time data generated during the test, a stress-strain curve was derived. Figure 3 shows the stress-strain curve for Test 22, which is typical in shape of the curves derived for each test. No correction was made for machine compliance, slack in the fixtures, or elastic deformation of the fixtures or inserts; therefore, the slope of the elastic portion of the curve shown in Fig. 3 does not equal Young's Modulus for the Zr-1Nb alloy.

The test number, specimen design type, temperature, strain rate, and mechanical properties for all tests are summarized in Table 2. The tests are grouped according to strain rate and temperature; within each group, the tests are listed in the order in which they were conducted. The specimen design types refer to the letters given in Table 1. Yield strength (YS) and ultimate tensile strength (UTS) were determined from the load vs. time strip chart. Uniform and total elongations were also measured on the strip chart and used to calculate uniform and total strains (percent elongations). By using the iterative process discussed in the Analysis section, the percent of the elongation that occurred in the gauge section was calculated; that percentage is listed in Table 2 as "% in Gauge." The percentage was multiplied by the uniform elongation, which was then compared to the original gauge length to determine the percent uniform elongation (UE). Finally, the ratio of total elongation to uniform elongation was calculated from the measured values from the strip chart and used to determine percent total elongation (TE), which is also given in Table 2.

Although the number of tests is fairly small, some definite conclusions can be drawn from the data presented in Table 2. Comparison of the room-temperature, low-strain-rate results from Tests 21-23 with those from Test 27 indicates consistency of the results independent of the specimen design type. The original purpose in the redesign effort was to find a specimen with a more uniform strain distribution across the width of the gauge section, and the small rise in the percent of elongation that takes place in the gauge section from 68 to74% (design A) to 76 to 80% (design B) is a result of that effort.

For Test 27, no total elongation value is given because that test was stopped at maximum load. As discussed below, the specimen was used to physically evaluate the percentage of strain that occurs in the gauge section, shoulder region, and outside the shoulder region.

Table 2. Summary of Results of Modified Ring Stretch Tests

Test	Specimen Design Type	Temp (°C)	Strain Rate (s^{-1})	YS (MPa)	UTS (MPa)	% in Gauge	UE (%)	TE (%)
6	A	25	0.001	194	402	74	13.2	68.4
7	A	25	0.001	183	377	68	10.1	56.3
21	B	25	0.001	217	329	78	16.1	58.9
22	B	25	0.001	221	332	76	19.8	61.2
23	B	25	0.001	224	331	80	18.3	52.7
27	A	25	0.001	223	333	74	18.4	[a]
Average[b]				221	331		18.2	57.6
Standard Deviation				3.1	1.5		1.5	4.4
15	A	25	1	360	420	[c]	12.9	51.5
16	A	25	1	289	409	[c]	12.9	51.5
17	A	25	1	311	387	[c]	17.2	38.6
Average				340	403		14.3	47.2
Standard Deviation				44	16.9			
9	A	400	0.001	92	192	67	13.2	69.2
24	B	400	0.001	110	183	78	18.3	74.5
25	B	400	0.001	111	195	78	14.5	69.7
26	B	400	0.001	103	171	78	15.4	57.3
Average				104	185		15.4	67.7
Standard Deviation				8.8	10.8		2.2	7.3
18	A	400	1	146	199	100	19.1	76.4
19	A	400	1	133	174	67	22.5	41.5
20	A	400	1	138	193	71	15.1	42.3
Average				139	189		18.9	53.4
Standard Deviation				6.6	13.1		3.7	19.9

[a]Test 27 was stopped at the maximum load; total elongation data were not available.
[b]Average and standard deviation do not include data from Tests 6 and 7.
[c]Percent strain in gauge section not calculated for these tests.

The results from Tests 6 and 7 are significantly different from the other four tests conducted at 25°C and a strain rate of 0.001 s^{-1}. Both the YS and UTS are much higher and the percent UE is lower for the two earlier tests; we can only conclude that these differences imply that insert lubrication was not as good in the earlier tests as in the later tests. The consistency of the other four tests, conducted at a different time, leads to the conclusion that the results from Test 6 and 7 should be ignored. A similar conclusion can be drawn for the three room-temperature tests conducted at the higher strain rate (Tests 15-17) because the average UTS for these three tests is so much higher than that for the lower-strain-rate tests at the same temperature, and the strain rate should not affect the UTS.

While it can be assumed that a similar lubrication problem existed with Tests 9, and 18-20, the UTS and percent UE of Test 9 and the UTS values of Tests 18-20 are statistically consistent with the values determined from Tests 24-26, which were also conducted at 400°C. The higher-temperature tests were apparently less affected by lubrication than the room-temperature tests, and this may be related to the different thermal expansions of the inserts (17-4 PH Stainless Steel) and the specimen. Regardless of the reason, the consistency of the data implies that the data from Tests 9, and 18-20 should be included in the discussions below, and the results from Test 9 were included in the averages given in Table 2 for the slow strain-rate, elevated-temperature tests.

Figure 4 shows the specimen after Test 22, which again is typical of the shape of the other specimens. All of the specimens broke on only one side, and the fracture across the gauge width was at approximately 45° from horizontal. The fractured gauge section had necked down before the break. As shown in Fig. 4b, the opposite gauge section also experienced some necking as well. Figure 4c is an edge view of the specimen; no significant thinning occurred during the test.

To physically confirm the high values of percent UE for Tests 21, 22, and 24, a series of microhardness indentations were placed around the circumference of the specimen for Test 27, and the specimen was pulled only until maximum load was achieved. The indentations were made with a Leitz microhardness tester and a small worm-gear device that turned the specimen in fairly uniform increments. The distance between the indentations was measured before and after the test (in the units of the machine). The purpose of this exercise was to show consistency between the actual strains occurring in the gauge section and those calculated through the iterative process discussed in the last section. The results of the measurements are summarized in Table 3, and indentations from one of the two gauge sections are shown before and after the test in Fig. 5. Figure 5a (before the test) shows indentations 6 through 10 from Series 1 and 1 through 4 from Series 3.

Table 3 gives the indentation series, an identifying number for each indentation, the original measurement, the final measurement, a location (outside the shoulder and gauge section, shoulder, or gauge) and the percent change. It should be understood that the measurements are from a numbered indentation to the next indentation. The first three sets of indentations were made through one gauge section, and the fourth set was made on the opposite side of the specimen.

Table 3. Results of Strain Measurements for Test 27

Series	Number	Original	New	Location	% change
1	1	76.4	76.7	Outside	0.4
	2	75	75	Outside	0.0
	3	75	75	Outside	0.0
	4	75	79.7	Shoulder	6.3
	5	70.3	78	Shoulder	11.0
	6	75	84.7	Shoulder	12.9
	7	75	89.3	Gauge	19.1
	8	75	91	Gauge	21.3
	9	75	92.2	Gauge	22.9
	10	75	82.4	Gauge	9.9
	11	78.5	84.7	Shoulder	7.9
	12	75	78.5	Shoulder	4.7
	13	77.5	78.4	Outside	1.2
	14	75	75.6	Outside	0.8
	15	77	77.8	Outside	1.0
	16	79.6	77.8	Outside	-2.3
	17	75	75	Outside	0.0
	18	75	75	Outside	0.0
	19	75	75	Outside	0.0
2	1	71	71	Outside	0.0
	2	75	75	Outside	0.0
	3	75	76.6	Shoulder	2.1
	4	75	78.4	Shoulder	4.5
	5	75	81	Shoulder	8.0
	6	75	93.6	Gauge	24.8
	7	75	87.5	Gauge	16.7
	8	75	89.3	Gauge	19.1
3	1	30.5	34.4	Gauge	12.8
	2	20	25	Gauge	25.0
	3	30.5	34.6	Gauge	13.4
4	1	78.5	81	Shoulder	3.2
	2	63.5	69.3	Shoulder	9.1
	3	80	89.3	Shoulder	11.6
	4	63.9	75	Gauge	17.4
	5	79.2	93.1	Gauge	17.6
	6	61.4	76.6	Gauge	24.8
	7	81.1	94.7	Gauge	16.8
	8	80.5	91.9	Gauge	14.2
	9	77.8	84.7	Shoulder	8.9

Several conclusions can be drawn from the data in Table 3. First, the data clearly show that a significant amount of strain occurs in the shoulder region, and that essentially no plastic strain occurs outside the shoulder and gauge regions. The latter observation confirms an assumption made in the Analysis section in discussing the iterative model. Average strain in the gauge section is 18.4%, with a standard deviation of 4.7, and a range of 9.9 to 25.0%. We can assume that if a similar exercise had been performed on a specimen with specimen design B, the standard deviation would be much lower because the strain distribution would be much more uniform. Uniform elongation measured from the strip chart was 0.45 mm. When compared to an original gauge length of 1.70 mm, the uniform strain would be 26.5% if all of the elongation occurred in the gauge section. By comparing the maximum uniform strain to that actually measured, we can conclude that only 72% of the elongation, on average, occurs in the gauge section. From the iterative process, we calculated that 74% of the elongation occurs in the gauge section, which agrees very well with the measured value. We can thus conclude that the simple model of using an iterative process to determine the percent of elongation that occurs in the gauge section is valid and can be used for the other tests.

Comparison of the averages from the tests conducted at 25°C to those at 400°C with a strain rate of 0.001 s^{-1} indicates that the temperature increase decreases strength, has essentially no effect on uniform ductility, and has very little effect on total elongation. As the temperature increases, the UTS drops by 44% and the YS drops by 37%. Such a drop in strength is expected with a rise in temperature. The percent UE shows a small drop as the temperature increases, but the drop is well within the data scatter; a drop in UE would not be expected with a rise in temperature. Finally, percent TE increases with the rise in temperature, but as with the drop in UE, the rise is within the scatter band of the data and may not be significant.

The effect of strain rate can be judged only by a comparison of the high-temperature tests because the results from the room-temperature, high-strain-rate tests are questionable. As the strain rate increased from 0.001 to 1 s^{-1}, the percent UE and YS increased, but the UTS remained constant. The small rise in uniform elongation may not be significant because the standard deviation from one dataset overlaps the uniform elongation data from the other. In addition, a drop in total elongation was seen as the strain rate increased. Typically, with an increase in strain rate, yield strength will increase (as was seen in these tests) and uniform and total elongations will both decrease. It is not clear why uniform elongation did not decrease with increasing strain rates, but the other two changes are consistent with expectations.

Discussion

Ideally, the plastic portion of the stress-strain curve up to the uniform tensile strength follows a power law. In that case, the work-hardening coefficient can be calculated from the yield strength (at ε_{YS}, typically 0.002), tensile strength, and uniform elongation by using the following relation:

$$n = \frac{\ln\left(\sigma_{YS} / \sigma_{UTS}\right)}{\ln\left(\varepsilon_{YS} / \varepsilon_{UE}\right)}, \qquad (1)$$

and the work hardening coefficient should be equal to the true uniform elongation, i.e., the true plastic strain at peak load.

An application of Eq. 1 to our data for Test 27 shows that n = 0.09, while true uniform elongation is 0.17 (the engineering uniform elongation given in Table 2 was 0.18). The small discrepancy between the work hardening coefficient and true strain at peak load can be explained as follows: Eq. 1 is derived from the Considère criterion, which relates the true stress (σ) and true plastic strain (ε) at peak load by:

$$\frac{d\sigma}{d\varepsilon} = \sigma \text{ or equivalently } \frac{\mathrm{d}\ln\sigma}{\mathrm{d}\ln\varepsilon} = \varepsilon \qquad (2)$$

The true strain at peak load being equal to Eq. 1 follows from Eq. 2 if the differences between true and engineering stress and strain can be ignored, and further, if a power law can be fitted to the stress-plastic strain curve from the yield strength to the onset of necking (i.e., the true stress equivalent to the ultimate tensile strength). For the VVER reactor material, however, a log-log plot of the true stress versus true plastic strain curve is nonlinear and cannot be fitted by a simple power law. Figure 6 shows a log-log plot of true stress vs. true plastic strain up to the peak load for Test 27; data points are plotted as open circles. Two lines are also shown in Fig. 6. The upper line (slope = 0.11) corresponds to the results from Eq. 1. The lower line with slope = 0.20 is tangent to the curve at the maximum load. The Considère criterion (Eq. 2) should still be valid for the Zr-1Nb cladding at the peak load. In other words, the slope of the log(σ) vs. log(ε) curve at peak load should equal the true plastic strain at peak load. The lower line in Fig. 6 has a slope of 0.20, which is fairly close to the true strain at peak load (indicated on the x-axis) and indicates that the Considère criterion is satisfied.

Conclusions

Based on the results and discussions given above, several conclusions can be drawn about the modified ring stretch tensile test and the mechanical properties of the Zr-1Nb alloy.

1. Comparison of early test results to more recent results points out the importance of eliminating friction between the inserts and the specimen. Friction leads to the appearance of higher strength values and lower ductility values.

2. Although some effort was made to redesign the specimen geometry, comparison of results from later tests conducted on both designs indicates that with good lubrication of the

inserts and proper analysis of the data, the results are consistent regardless of specimen geometry.

3. Uniform elongation for the few tests conducted was independent of temperature in the range of 25 to 400°C. A small drop was actually seen as the temperature increased, but was within the scatter of the data.

4. A small increase in total elongation was noted as the temperature increased to 400°C but the small number of tests conducted made it difficult to conclude if the rise is significant.

5. Both ultimate tensile strength and yield strength decline ≈40% as the temperature increases from 25 to 400°C.

6. For an increase in strain rate from 0.001 to 1 s^{-1}, yield strength and uniform elongation increase by ≈34%, while total elongation drops by almost the same percentage.

References

1. S. Arsene and J. Bai, "A New Approach to Measuring Transverse Properties of Structural Tubing by a Ring Test," J. Testing Evaluation, **24** [6], Nov. 1996, pp. 386-391.

2. B. Josefsson and V. Grigoriev, "Modified Ring Tensile Testing and a New Method for Fracture Toughness Testing of Irradiated Cladding," Studsvik Material AB, S-611 82 Nyköping, Sweden, May 22, 1996.

Fig. 1. (a) gives a Schematic diagram of three-part tooling inserted into specimen for modified ring stretch tensile test.

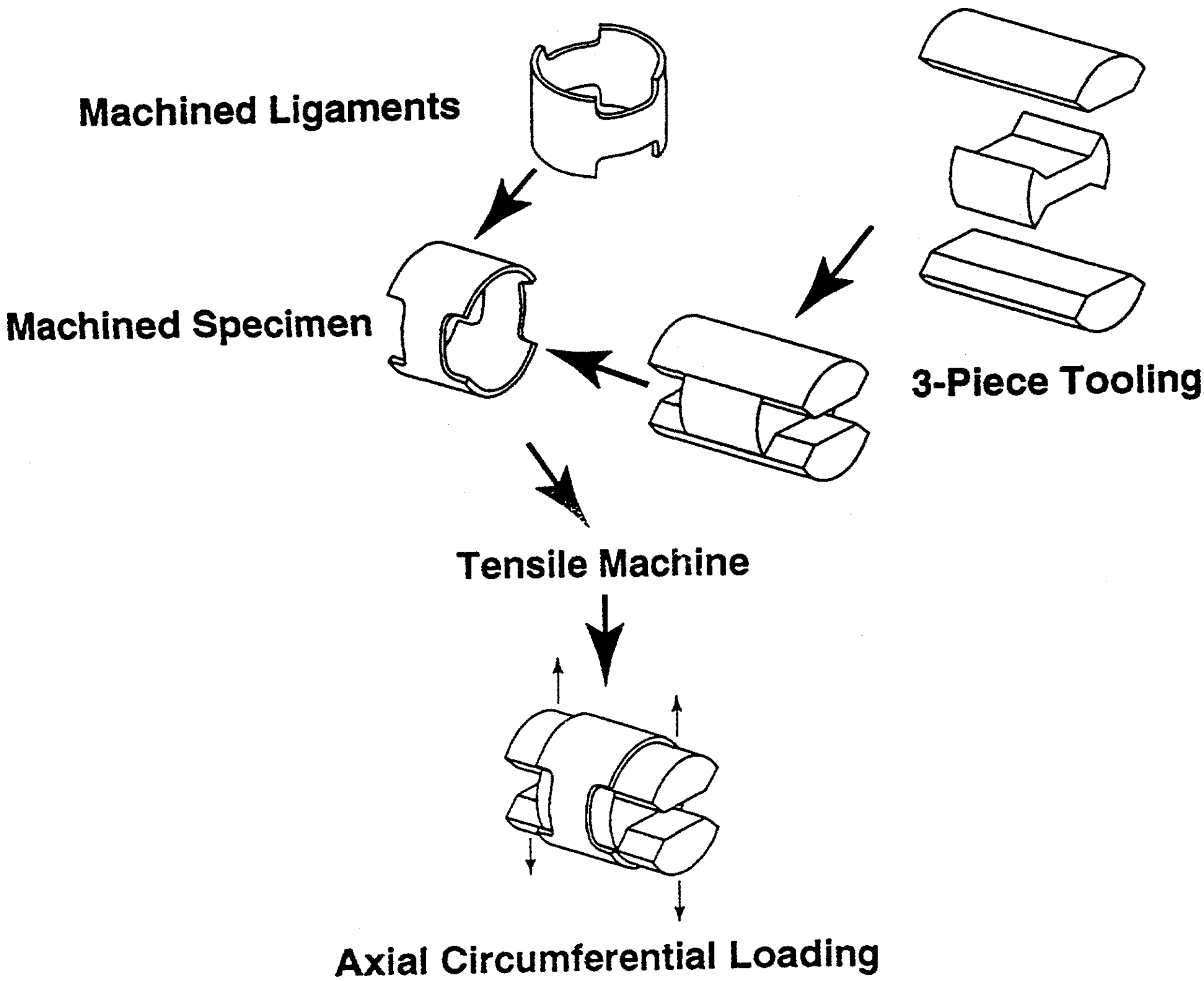


Fig. 1. (cont'd) (b) The two fixtures (with the machined threads) used for pulling on the inserts; inserts, central spacer, and post-test specimen are also shown. (c) Bottom view of fixtures with inserts partly inserted.

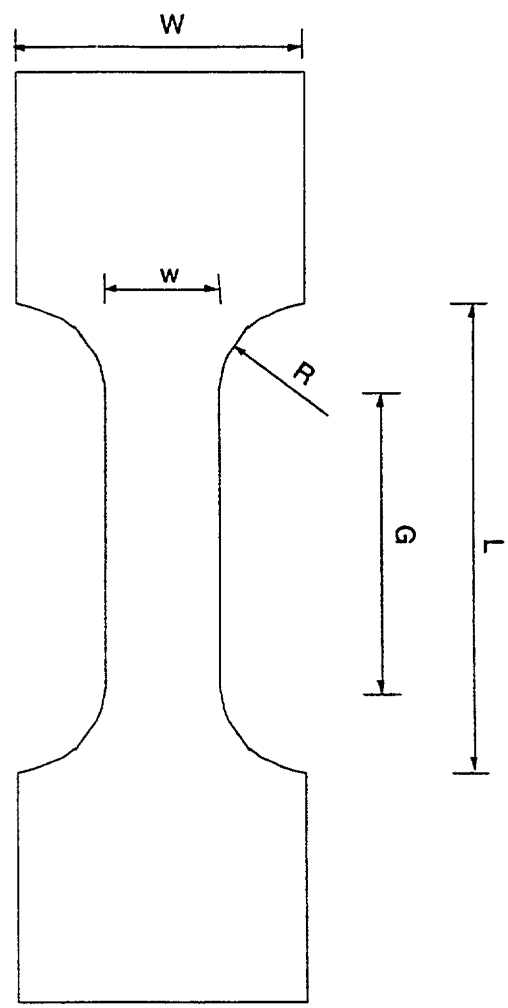
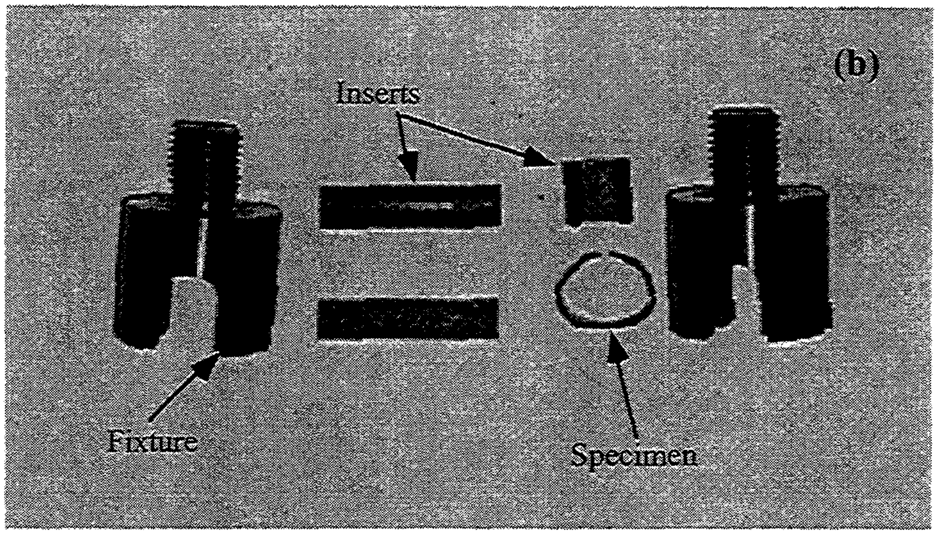


Fig. 2: Schematic diagram of gauge section from modified ring stretch specimen. Although L and G indicate measures of chord, it is the arc measure that is used for strain calculations.



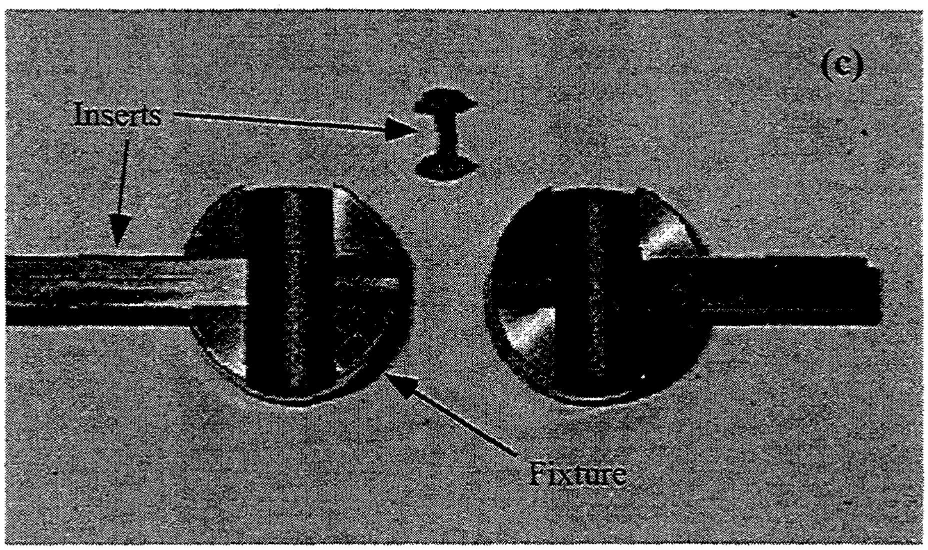


Fig. 3: Stress-strain curve for Test 22; line for determining uniform elongation is shown parallel to elastic portion of curve.

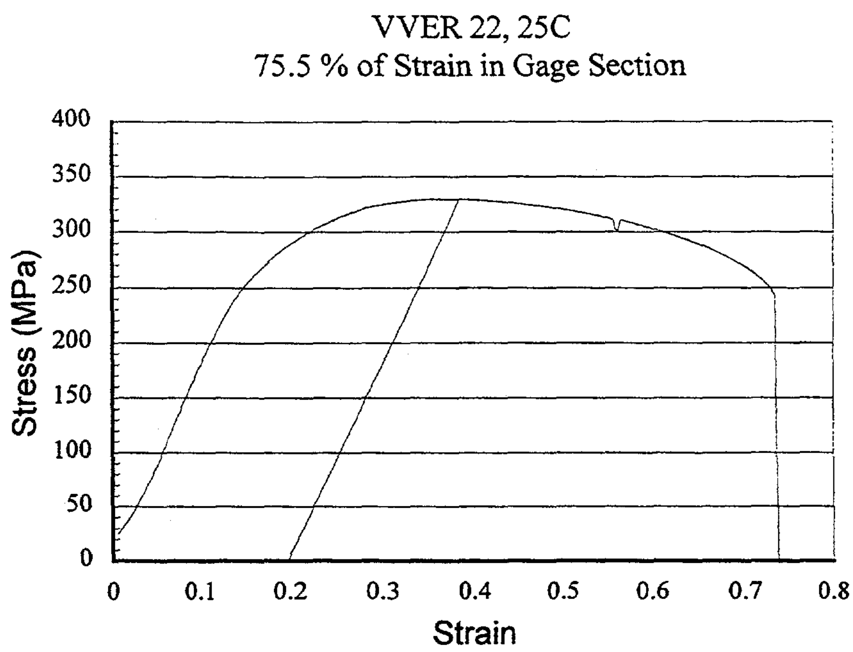


Fig. 4: Post-test condition of specimen from Test 22; the scale to the right is in millimeters. (a) shows the gage section that fractured (14x), (b) shows the opposite gage section that necked down but didn't fracture, and (c) shows the edge view of the specimen.

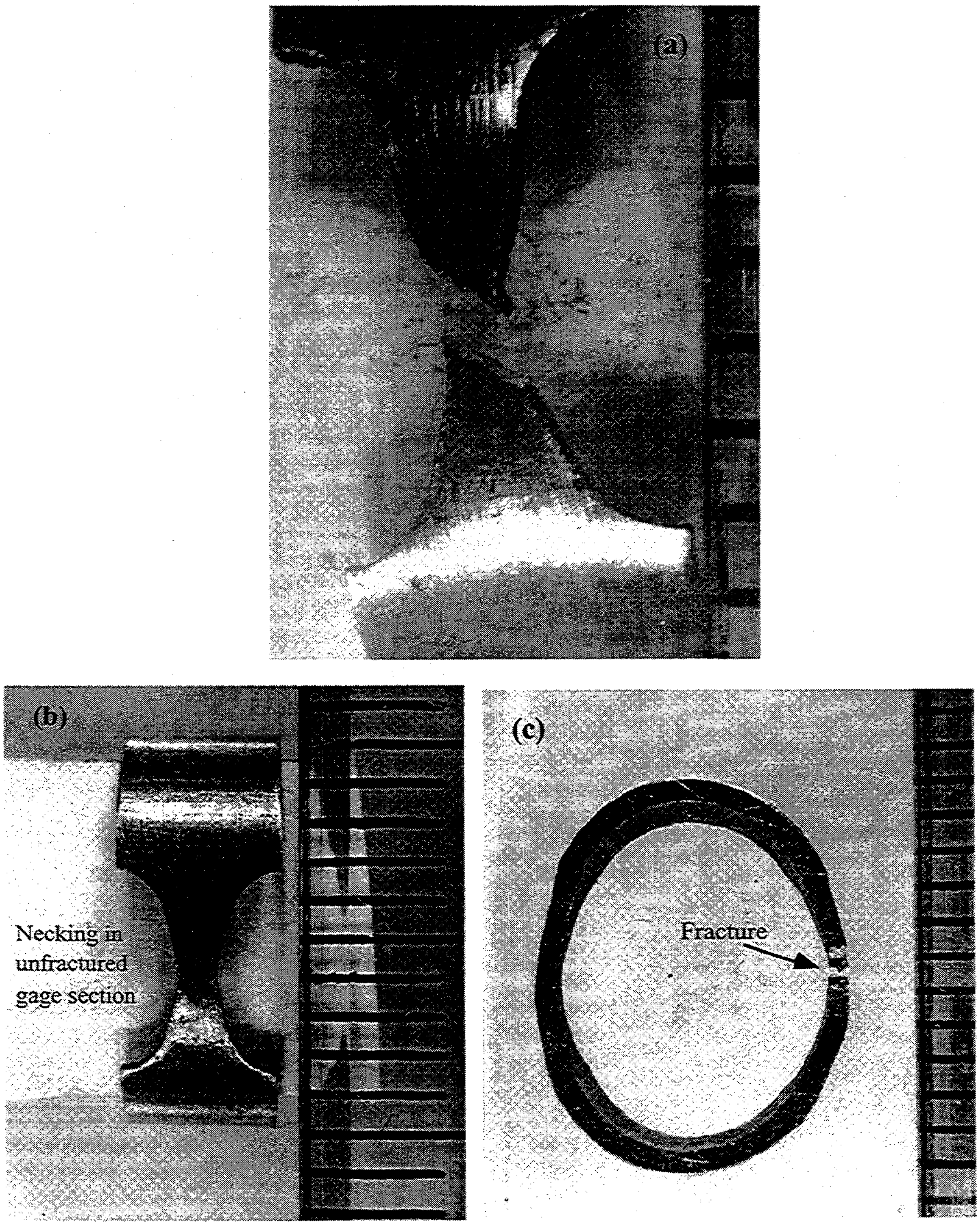


Fig. 5: Microhardness indentations on one gage section before (a) and after (b) Test 27. 75x.

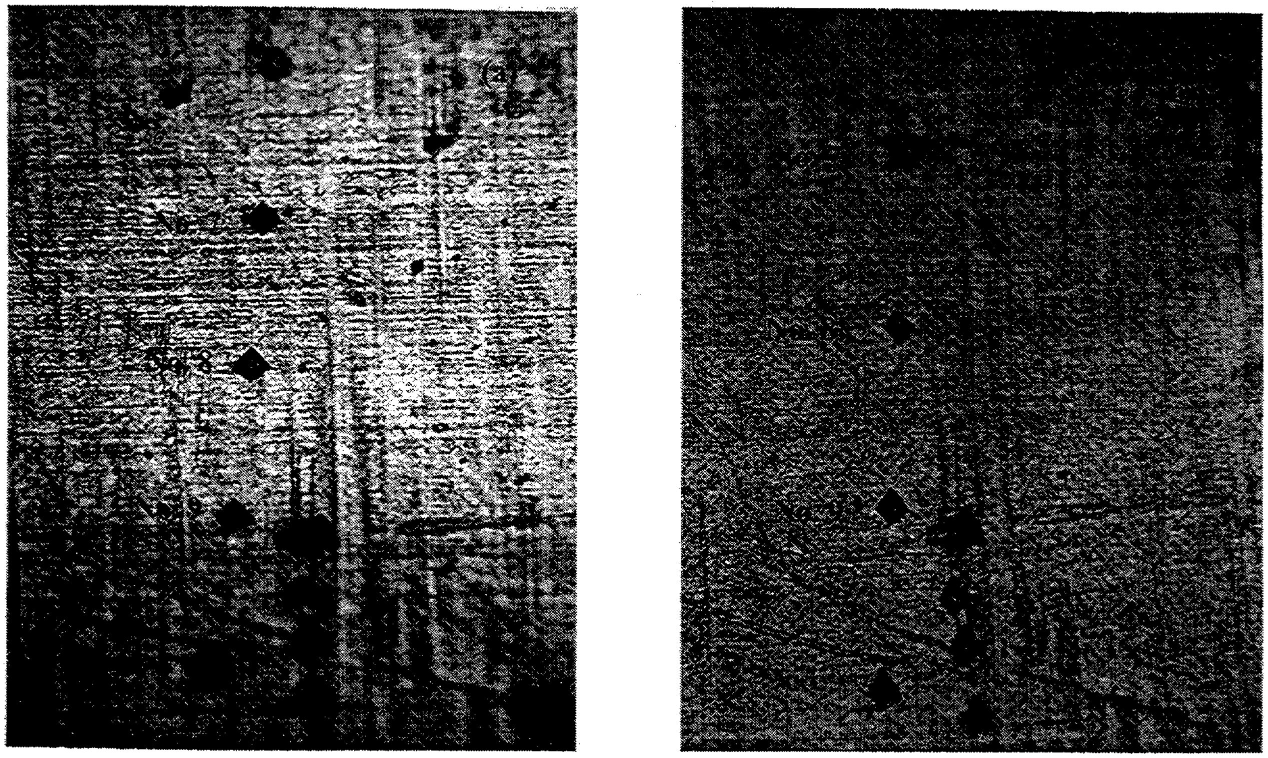
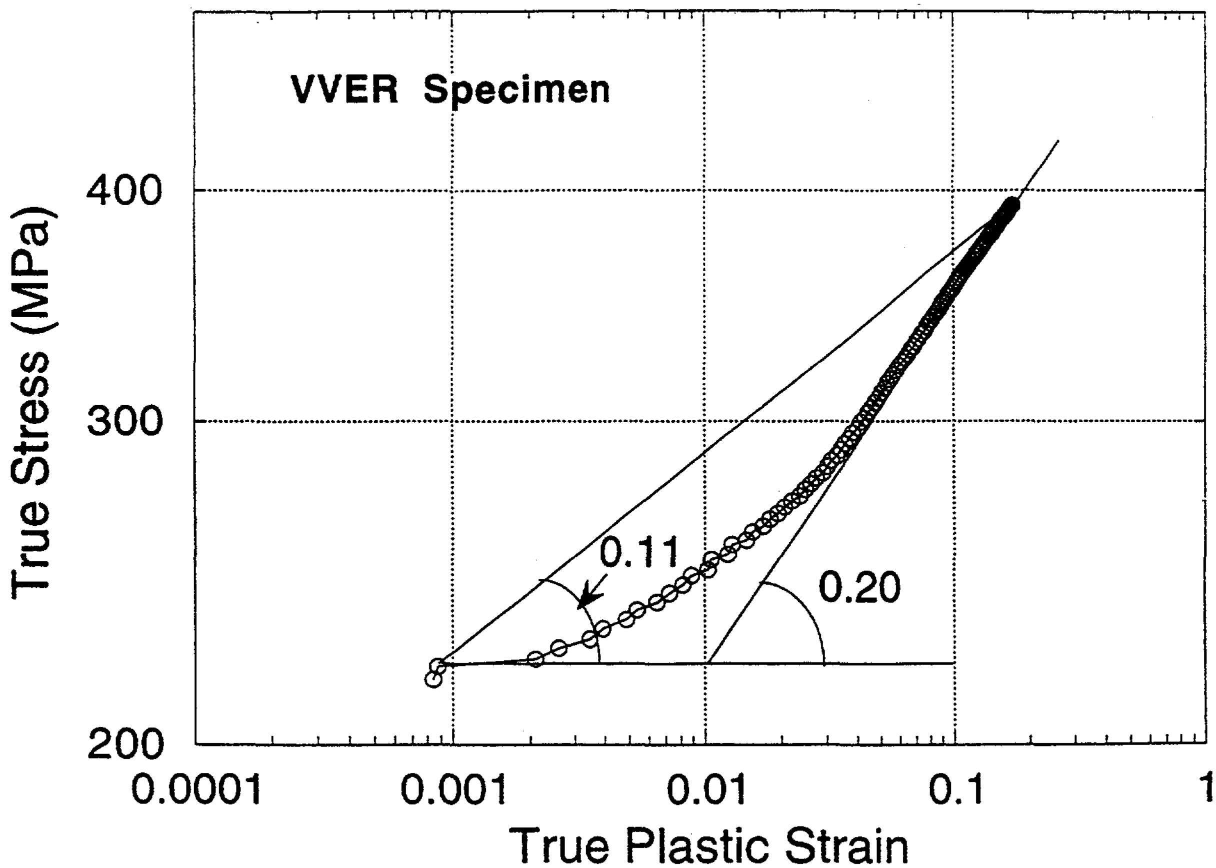


Fig. 6. Log-log plot of true stress vs. true plastic strain for Test 27 up to point of necking; plastic stress-strain does not follow power law relation.



THE INFLUENCE OF STRAIN RATE AND HYDROGEN ON THE PLANE-STRAIN DUCTILITY OF ZIRCALOY CLADDING

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Abstract

We have studied the ductility of unirradiated Zircaloy-4 cladding under loading conditions prototypical of those found in reactivity-initiated accidents (RIA), i.e.: near plane-strain deformation in the hoop direction (transverse to the cladding axis) at room temperature and 300°C and high strain rates. To conduct these studies, we developed a specimen configuration in which near plane-strain deformation is achieved in the gage section, and a testing methodology that allows us to determine both the limit strain at the onset of localized necking and the fracture strain. Our experiments indicate that there is little effect of strain rate (10^{-3} to 10^{2} s^{-1}) on the ductility of unhydrided Zircaloy tubing deformed under near plane-strain conditions at either room temperature or 300°C. Preliminary experiments on cladding containing 190 ppm hydrogen show only a small loss of fracture strain but no clear effect on limit strain. Our experiments also indicate that there is a significant loss of Zircaloy ductility when surface flaws are present in the form of thickness imperfections.

Introduction

During a reactivity-initiated accident (RIA), a control rod ejection or drop causes a sudden increase in reactor power, which deposits a large amount of energy in the fuel. This energy deposition causes the fuel to expand and fission gas contained in the fuel to be released. Both of these factors cause loading to the cladding. The energy deposition limits to avoid cladding failure and fuel dispersal were originally determined based on tests conducted on fresh cladding and on cladding irradiated up to 30 GWd/t [1]. Recent results have suggested that the ability of the cladding to withstand an RIA may be degraded after long exposure to the reactor environment [2-4].

Predicting the cladding ductility (and survivability) during a RIA event is further complicated by the likelihood that the cladding is subjected to a range of deformation paths during fuel expansion/fission gas evolution; the cladding is not subjected simply to uniaxial tension, as is monitored in a convention ring test. For thin-wall cladding deforming under plane stress conditions due to through-thickness slip, it is well known from the sheet metal forming literature that the plane-strain deformation path is particularly severe in limiting ductility. In this deformation path, at least locally along the cladding, hoop expansion occurs with little or no axial extension of the cladding. Under these conditions, the cladding ductility can be significantly less than that measure under uniaxial tension conditions. For example, theory predicts that plane-strain ductility is roughly 50% of uniaxial tension ductility [5]. Experiments on recrystallized Zircaloy 2 show an even greater disparity of ductility between plane-strain and uniaxial tension [6]. Thus, the use of failure criteria based primarily on uniaxial ring tests for predicting the performance of cladding during a RIA is questionable.

The purpose of this research program is to investigate the deformation and fracture of unirradiated Zircaloy 4 cladding subjected to following RIA-like loading conditions: transverse (hoop) extension of the cladding under near plane-strain conditions at both room temperature and at 300°C as well as under both quasi-static and dynamic loading conditions. Preliminary results for the influence of hydrogen are also presented. In order to obtain a near plane-strain deformation path in a cladding specimen, we have used both experiment and finite element analysis to design a new straightforward test which subjects the cladding to transverse extension under conditions in which there is little axial extension. We denote this test geometry as the "transverse plane-strain tension test." This communication describes both the test procedure and the influence of test temperature, strain rate, and hydrogen (at the 190 ppm level) on the transverse plane-strain ductility of unirradiated Zircaloy 4 cladding. We also show here the results of a study of the influence of small thickness imperfections on Zircaloy ductility under plane-strain conditions.

2. Design of transverse plane-strain tension test

A basic premise for this study is that failure initiation of Zircaloy cladding, if it occurs during an RIA event, is most likely to occur at least locally due to hoop expansion of the cladding with little or no axial extension. In such a case, cladding failure is characterized by fracture along the tube axis under conditions in which a "transverse plane-strain deformation path" causes failure in the form of a crack, which then propagates along the tube axis. In support of this assumption, long axial cracks have been observed in RIA tests conducted in the CABRI and NSRR facilities [2, 4]. To obtain measures of cladding performance under these conditions, new experimental procedures need to be designed in order to accurately monitor those material properties which potentially limit failure of cladding in service. To this end, we have designed a straightforward, transverse plane-strain tension test. The behavior of the cladding subjected to this test will be contrasted to that in a more conventional ring test in which the material undergoes uniaxial tension during deformation.

Specimen geometries for both uniaxial tension and plane-strain deformation were designed to permit transverse tensile testing of 0.95 cm (0.375 inch) outer diameter Zircaloy-4 cladding. For uniaxial tension testing, thin-ring specimens with two gage sections 0.64 cm (0.250 inches) long and 1.0 mm (0.040 inches) wide were prepared by wire EDM machining. This specimen geometry is shown in Figure 1.

For transverse plane-strain testing, edge-notched specimens with two gage sections, each with a 0.20 cm (0.080 inch) notch diameter and a 0.635 cm (0.250 inch) ligament width were machined. Figure 2 shows the geometry of this specimen, which resulted from an evaluation of six different specimen geometries. Specimens with notch diameters ranging from 0.2 cm to 0.31 cm (0.080-0.125 inches) for a fixed ligament width of 0.635 cm (0.250 inches) and ligament widths ranging from 0.51-0.76 cm (0.200-0.300 inches) for a fixed notch diameter of 0.2 cm (0.080 inches) were examined experimentally for the degree of plane-strain behavior. In these tests, both the minor and major strain distributions were measured within the deforming gage section at several interruptions during the tests and after failure. On the basis of minimizing the ratio of minor strain to major strain, the specimen with 0.2 cm (0.080 inch) notch diameter and a 0.635 cm (0.250 inch) ligament width was selected. As shown in Figure 3, this specimen induces near plane-strain deformation behavior of the central 40% of the gage section.

Finite element modeling (FEM) was also used to evaluate the same six specimen geometries mentioned above. FEM was performed using Abaqus version 5.4 to run the calculations and FEMAP version 4.1 for mesh generation and post-processing of the results. We used the experimentally determined stress-strain response of the cladding as determined from transverse compression testing[7] in the FEM calculations. For simplicity, flat 2-D versions of the notch geometries were modeled. Consistent with experimental results, the FEM predictions also show plane-strain deformation within the central section of the gage width. However, as shown also in Figure 3, the FEM results suggest that the region of plane-strain is expanded to about 60% of the gage width, compared to the 40% observed. The experimental observations also indicate a finite minor strain at the center of the specimen, while the FEM predict near-zero minor strain; see Figure 3. We believe that the discrepancy between FEM predictions and experimental observations results at least in part from the assumption of isotropic plasticity in the FEM code and the fact that the Zircaloy is plastically anisotropic. Nevertheless, Figure 3 shows reasonable agreement between the trends of the data.

Finally, we note that it is important to realize that failure initiation occurs within the center section of the specimen (near a plane-strain condition) and propagates to the outer edges of the specimen (which are subjected to a local deformation path near uniaxial tension). Thus, a determination of the failure condition needs to be performed on a local strain basis in which failure strains are determined near the specimen center.

3. Transverse plane-strain tension: experimental procedure

The transverse tensile testing of the cold worked, stress relieved Zircaloy 4 cladding was performed using an Instron 4206 mechanical test frame using sets of die inserts and a pin-loaded grip assembly, machined from hardened 17-4 precipitation-hardenable stainless steel, to load the specimen. A schematic diagram of this loading assembly is shown in Figure 4. For testing, the gage sections of the specimens are positioned at the top and bottom positions of the die inserts (not at the opening between the inserts) in order to minimize bending strains due to straightening of the cladding walls during deformation. In this orientation, the gage sections of the cladding are maintained at the constant curvature of the die inserts during deformation. In order to minimize friction between the die inserts and the inner wall of the cladding specimen, we lubricated the cladding-inserts interface with 2 layers of vacuum grease and Teflon tape at the beginning of each test.

During testing, local strains are measured directly from the specimens by a process of microhardness gridding. In this process, an array of microhardness indents is applied to the samples before testing. The distances between indents are then measured using a traveling microscope before testing, at several interruptions during the tests if necessary, and after failure. True local strains can then be calculated directly from these measurements. For hydrogen testing, samples were charged with hydrogen in a furnace where they were held at 400°C for 24h under a prescribed hydrogen pressure. Examination of the hydrided samples showed that the hydrogen was in the form of circumferential hydrides, homogeneously distributed through the specimen thickness. The level of hydrogen was measured by hot vacuum extraction, by Luvak, Inc..

4. Effect of temperature on transverse plane-strain ductility

The transverse plane-strain ductility of Zircaloy cladding was measured under quasi-static deformation conditions at both 25°C and 300°C; a minimum of three tests per condition were conducted. Examination of the fracture surfaces shows that the failure initiates in the center of the specimen and propagates outward to the edges of the sample on a plane approximately 45° through the cladding thickness. Figure 5a shows the major strain distributions at one interruption and at failure along the gage length of the specimen at room temperature. There are two sets of data for each interval, because two rows of microhardness indents are put on the sample and measured. This figure shows the evolution of strain from uniform deformation at small strains (first interruption) to the development of localized necking at larger strains. From Figure 5a, we may define two measures of material ductility: (a) the "limit" strain and (b) the fracture strain. The limit strain defines failure at the onset of localized necking, which is close to the ductility of a very long tensile specimen which fails due to localized necking. The fracture strain is simply the maximum strain sustained within the gage section. For the data in Figure 5a, the limit strain is ε_L=0.09. For cladding extension beyond the limit strain, deformation quickly localizes into a strip of material approximately 1 mm wide, which constitutes the localized neck shown in Figure 5a. Specimen fracture then occurs at a local fracture strain of about 0.23, as shown in Figure 5a.

Quasi-static transverse plane-strain tests have also been carried out at 300° C, and Figure 5b shows the corresponding major strain distribution along the gage length at failure. It is evident that the true local fracture strain within localized neck is much higher at 300° C than at room temperature. However, the limit strain data show no significant effect of temperature; the limit strains observed at 300°C are similar to those at room temperature. This effect is consistent with the dependence of the limit strain on the strain hardening and strain-rate hardening of the cladding [8] and the fact that these parameters do not differ much between these two test temperatures [9].

5. Cladding ductility and surface flaws

To examine the sensitivity of cladding ductility to surface flaws, slow strain-rate tests have been performed on notched plane-strain specimens containing shallow grooves across the entire width of the outer surface of the gage section. These grooves, which form thickness imperfections spanning the gage width, are approximately 0.3 mm wide and range from 10 to 80 microns deep. Tests of these grooved specimens have been performed at 25 and 300° C. Failure in these specimens is similar to the smooth, ungrooved specimens in that failure initiates in the center of the specimen and occurs on a plane also approximately 45° through the cladding thickness. The localized neck is contained almost entirely within the groove, while the regions outside of the groove experience nearly uniform deformation. As such, the failure strain is measured as an average of the strain values accumulated outside of the groove.

As shown in Figure 6 for room temperature tests, cladding ductility is quite sensitive to the presence of surface flaws in the form of the grooves or thickness imperfections. Furthermore, the experimental results obtained from the grooved specimens can be directly compared to the theoretical predictions of our previous localized necking analysis [10]. In this figure, the limit strain is plotted against the imperfection severity, *f*, (the ratio of imperfection depth to cladding thickness). Figure 6 shows very good agreement between experiment and theory, which supports our predictions of the surface-flaw sensitivity of Zircaloy ductility. It should be noted that five ungrooved test specimen results have been included and compared with predictions based on an imperfection severity of 0.01. We have measured the thickness variation around the perimeter of the cladding and found up to 3% variation in thickness. Thus, we believe that it is safe to assume that a 1% thickness imperfection can exist within the notched gage section of the as-received cladding simply due to the cladding fabrication process.

6. Plane-Strain Tension vs. Uniaxial Tension Behavior

Quasi-static room temperature tests were performed on uniaxial ring specimens in addition to the notched plane-strain specimens. Figure 7 contrasts a failed uniaxial ring specimen with a double notch plane-strain specimen. As this figure shows, failure occurs due to different modes of slip in these two specimens. As discussed before, the notched plane-strain specimens fail due to through-thickness slip occurring on a plane inclined approximately 45° through the cladding thickness, which is consistent with orientation of dominant failure plane observed in the failures of high burn up cladding tested in Japan [4]. This behavior is consistent with our premise the cladding failure due to hoop expansion must occur due to predominantly through-thickness slip. On the other hand, the uniaxial ring specimens primarily fail due to slip roughly inclined 45° across the <u>width</u> of the sample due to the plastic anisotropy of the cladding which renders through-thickness slip difficult. Ductile failure of cladding due to this slip mode is very unlikely under RIA loading conditions. Thus, we conclude that failure of the two specimen geometries is a result of differing modes of slip; in particular, the uniaxial ring test fails in a manner inconsistent with cladding failure. In contrast, failure initiation in the transverse plane-strain specimen occurs as a result of through-thickness slip along an axial path in a manner consistent with that expected during an RIA incident.

7. Plane Strain Testing

Using the geometry shown in figures 2 and 4, we conducted plane strain testing of Zircaloy cladding, under the following combinations: room temperature and 300°C, 10^{-3} /s and 10^{2}/s, and 40 and 190 ppm H. We measured both limit strain and fracture strain; these results are shown in figures 8 and 9. It should be pointed out that there is greater uncertainty in the measurement of limit strain, stemming from the fact that the measurement needs to be done on the "shoulder" of the localized neck (see figure 5). Experimentally, the location of the shoulder is often near the end of the gage section, and this introduces uncertainty into limit strain measurements.

<u>Effect of temperature :</u> Because of increased plasticity at higher temperature, we expect to measure higher strains at 300°C than at room temperature. This was in fact the case, as the fracture strain at 300°C was about 50% higher than at room temperature, for all conditions. The limit strain at 300°C was not much different than at room temperature at the lower strain rate, but at the high strain rate, the limit strain was considerably higher at 300°C than at room temperature.

<u>Effect of strain rate:</u> The influence of strain rate on failure of the Zircaloy cladding was assessed by examining the fracture strain at a strain rate of 10^{-3}/s with that obtained after deformation at 10^{2}/s. As shown in Figure 8, there is little effect of strain rate on the fracture strain at either room temperature or 300° C. In contrast, preliminary analysis of data from unhydrided cladding show that the limit strain decreases slightly (from 0.08 to 0.05) upon increasing strain rate at room temperature, but it appears to increase somewhat (from 0.08 to 0.10) at 300° C at the higher strain rate.

<u>Effect of hydrogen</u>: Figures 8 and 9 also show that hydrogen at the level and distribution examined here have a relatively minor effect on cladding ductility. The higher hydrogen content resulted in lower fracture strains for all conditions studied. The differences, however, are close to our experimental accuracy. The data on limit strain is more complex. At room temperature there appears to be a slight decrease in the limit strain with strain rate, although the difference may not be statistically significant. At 300 °C the limit strain is higher at the higher strain rate, especially in the sample with higher hydrogen

content. This preliminary limit strain data suggest that there may be a slight effect of the hydrides, decreasing the limit strains at low strain rates but increasing them at high strain rates. However, the magnitude of the effects are on the scale of the experimental scatter within the data. Thus, we must conclude that there is no statistically significant effect of hydrides at the 190 wt. ppm level on the limit strain in these near plane-strain tests.

We should note that the hydrogen levels used here are not at the upper limit of the possible levels found in a reactor. Moreover, the hydrides were distributed homogeneously over the sample. If the hydrides were instead localized in blisters, or in a hydrided rim, (as they are commonly found in-reactor) they would be more likely to affect Zircaloy ductility. Tests based on cladding where there is hydrogen localization are in progress.

Conclusions

We have studied the ductility of Zircaloy 4 cladding under loading conditions relevant to an RIA. Based on unirradiated material tested at 25°C and 300°C as well as at strain rates of 10^{-3} and 10^{2}/s, the principal results of the study are as follows:

1. A straightforward, "transverse plane-strain" specimen has been designed. Both finite element analysis and experiments have been used to optimize the specimen configuration. This specimen subjects the cladding to loading in the hoop direction, transverse to the cladding axis, but uses notches to constrain the deformation path within the gage section to a condition of near plane strain along the cladding axis. A gridding technique has been employed to determine two cladding failure conditions: (a) a limit strain which indicates the onset of a localized necking failure and (b) a fracture strain which indicates the local strain across the fracture surface.
2. The cladding ductility, as measured by the limit strain, is very sensitive to the presence of surface flaws in the form of thickness imperfections across the gage section. The present experimental results are consistent with an earlier theoretical analysis [10].
3. A comparison of cladding failure in our transverse plane-strain tension test with that in a uniaxial ring test indicates a difference in failure paths with the ring test failing on an inclined plane across the specimen width while the plane-strain specimen is forced to fail as a result of through-thickness slip along the cladding axis, as in an RIA event. This indicates that care should be taken in using the results of uniaxial ring tests to assess Zircaloy ductility.
4. At 300°C, the cladding exhibits a higher fracture strain than at room temperature for each of the conditions tested. The limit strain increases significantly at 300°C compared to 25°C for the high strain rate test, but does not change when tested at 25 and 300°C at low strain rate.
5. There is a minimal influence of strain rate (10^{-3}/s vs. 10^{2}/s) on the ductility of unhydrided Zircaloy cladding at either 25 or 300°C. While there is no significant effect on fracture strain data, the limit strain decreases slightly at room temperature but increases at the elevated temperature.
6. Increased hydrogen content slightly decreases the fracture strain for the conditions studied. There is no statistically significant effect of hydrides at the 190 ppm level (and uniformly distributed) on the limit strain in any of the test temperature/stain rate conditions examined.

Acknowledgments

We are indebted to Ross Bradley of Sandvik Metal for supplying the samples for this study, and to Douglas Bates for help with conducting the experiments. We also thank Charles Roe of the Naval

Surface Warfare Center at Carderock for conducting the high strain experiments. Finally we thank Ralph Meyer for his encouragement and support of this research. This research was funded by the Nuclear Regulatory Commission under Educational Research Grant NRC-04-95-068.

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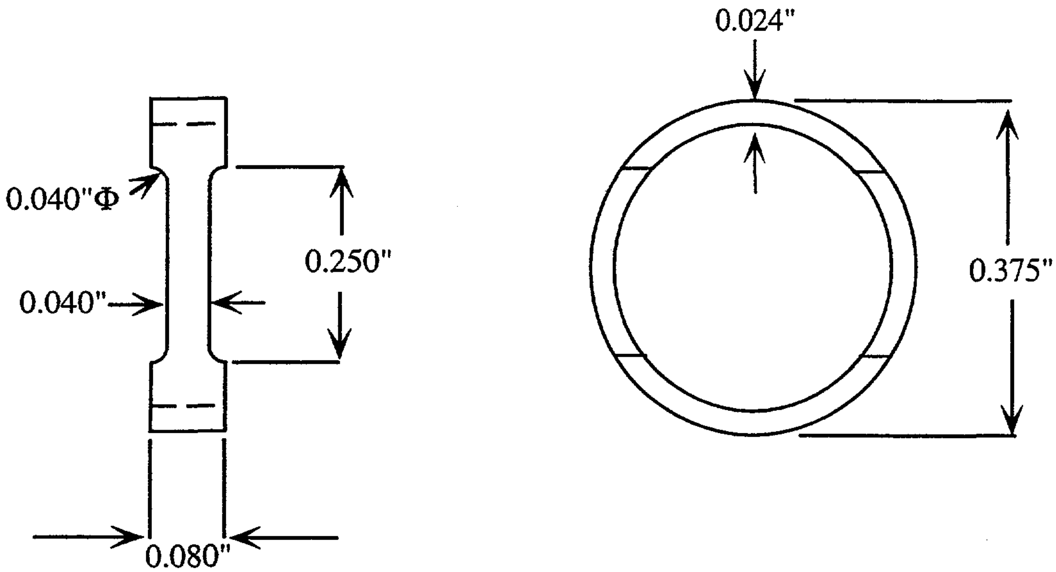


Figure 1: Geometry of test specimen designed for uniaxial tension testing in the transverse direction.

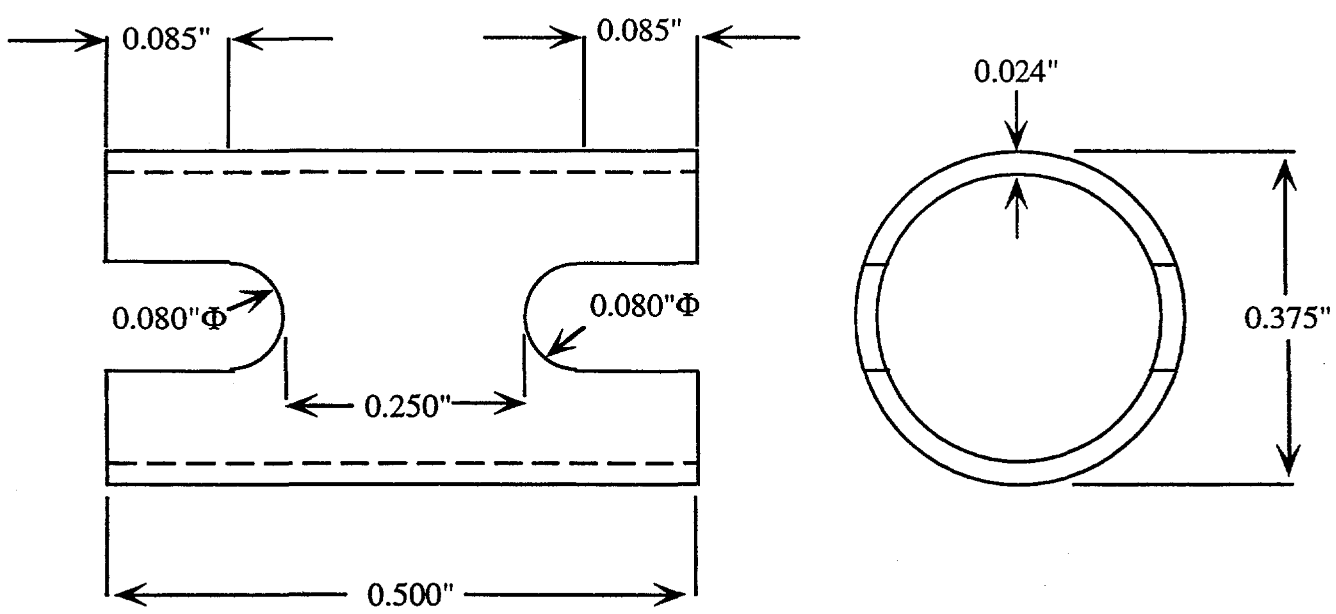


Figure 2: Geometry of test specimen designed for plane-strain tension testing in the transverse direction.

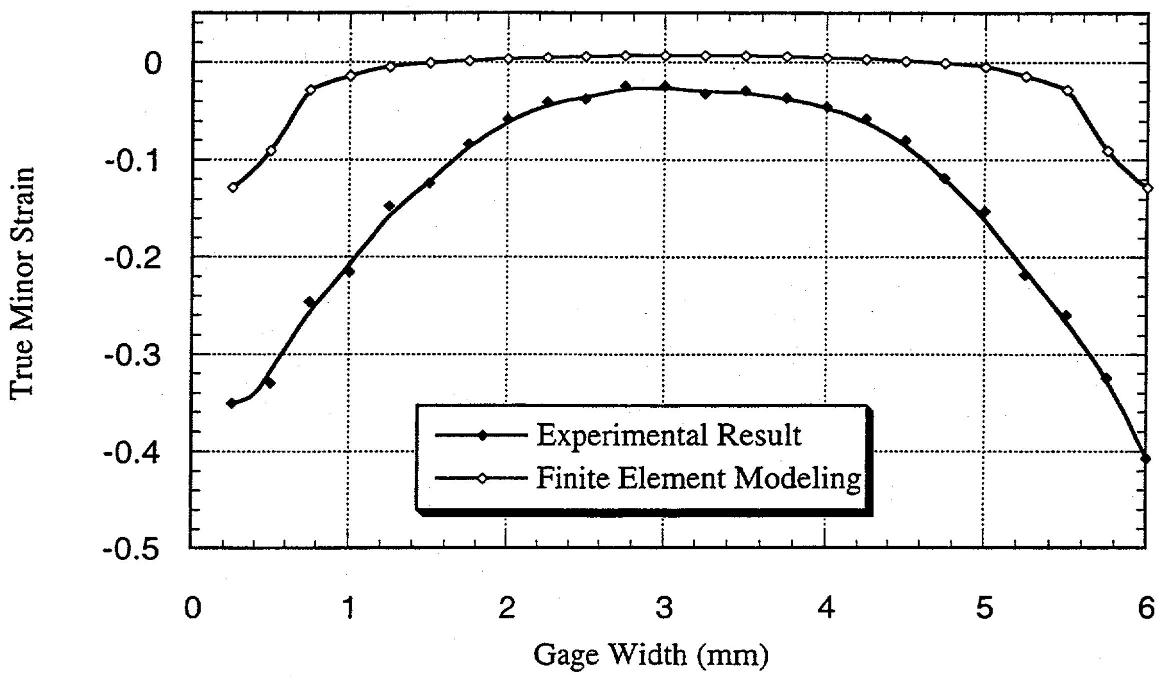


Figure 3: Experimental measurements and FEM predictions of width strain as a function of gage width for notched plane-strain specimen at room temperature.

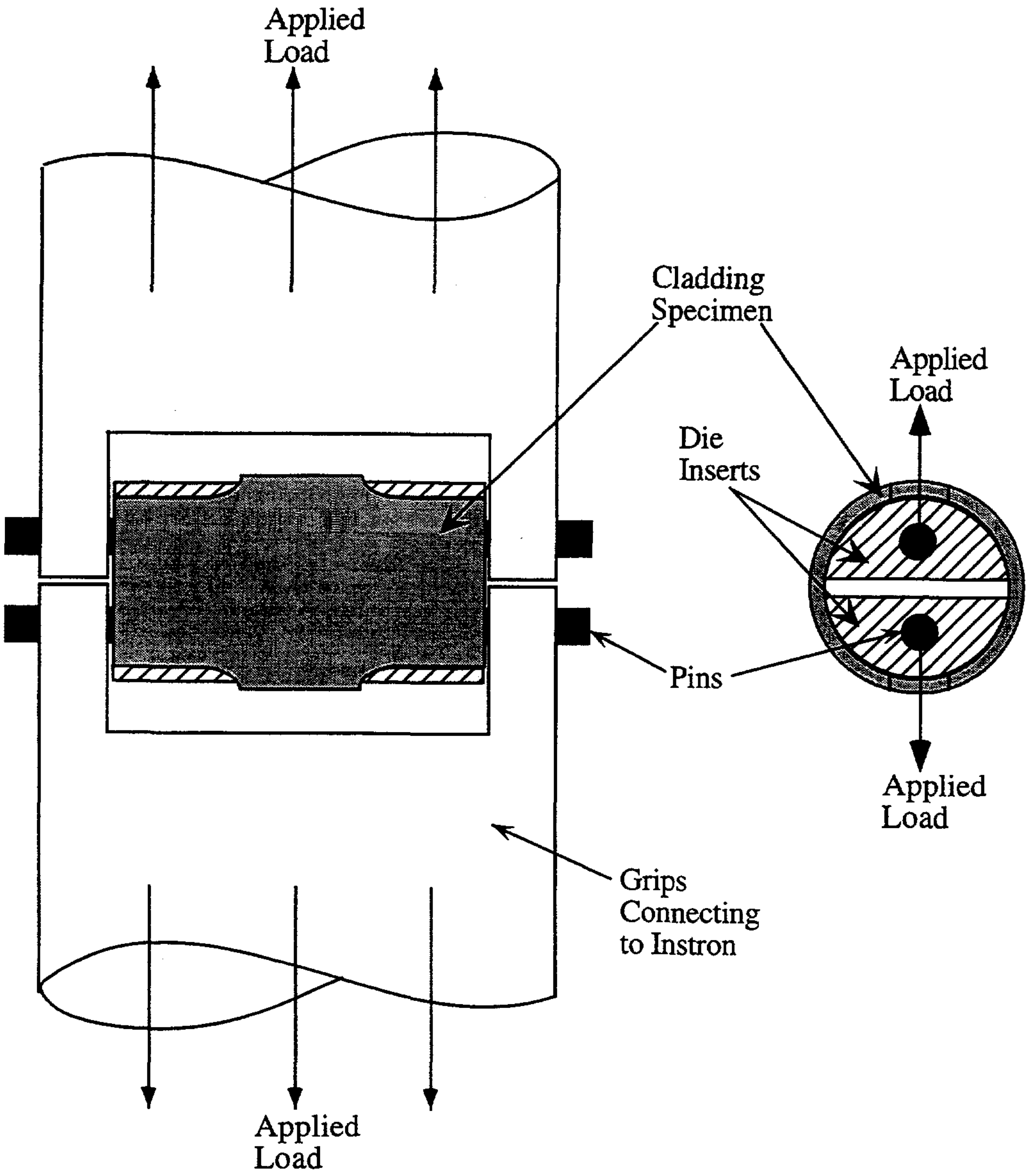
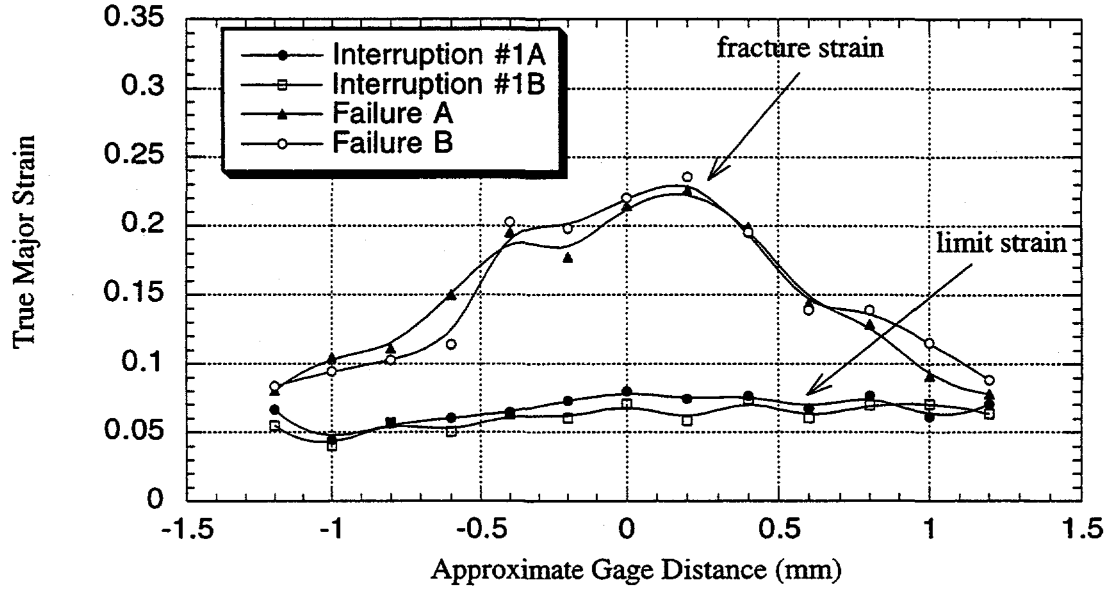
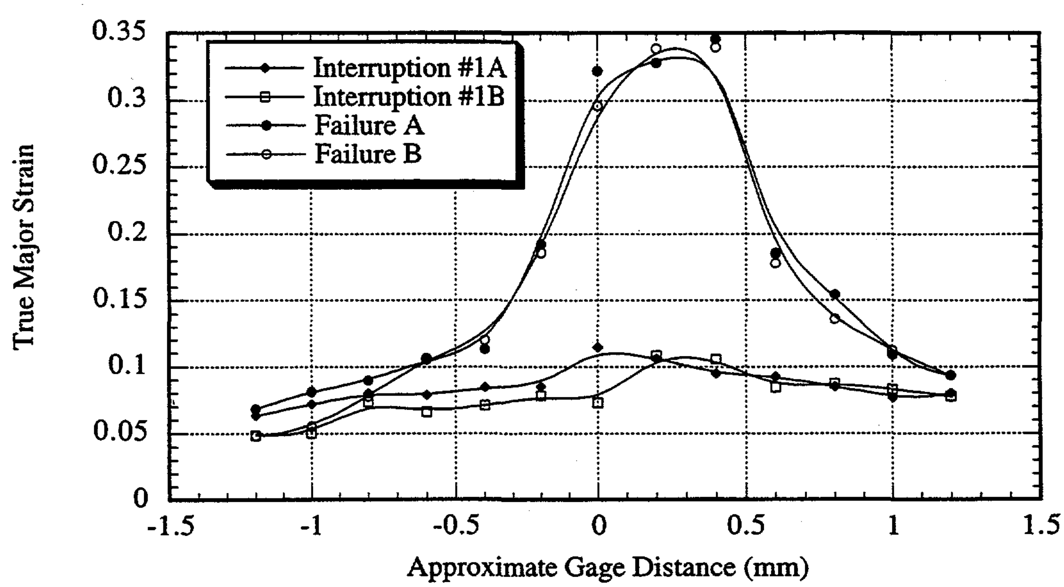


Figure 4: Schematic representation of loading assembly designed for transverse tensile testing of cladding specimens.



(a)



(b)

Figure 5: Major strain distribution at one interruption and at fracture as a function of gage length distance for (a) 25 C and (b) 300 C under quasi-static, plane-strain loading.

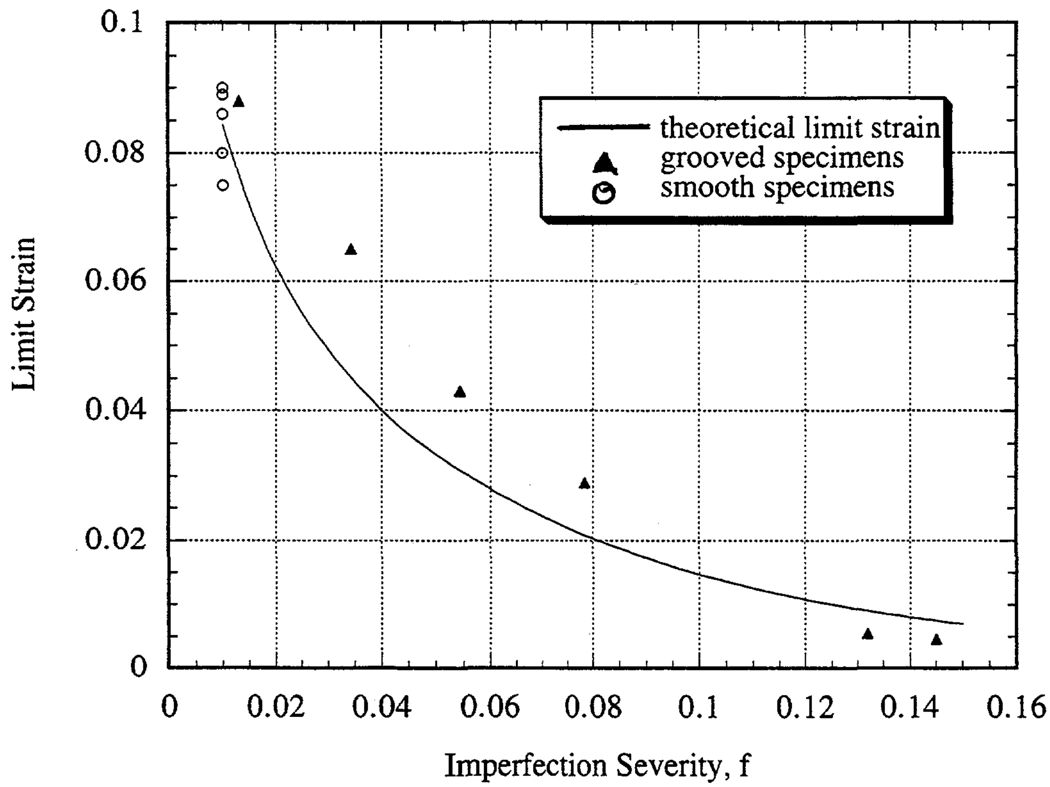
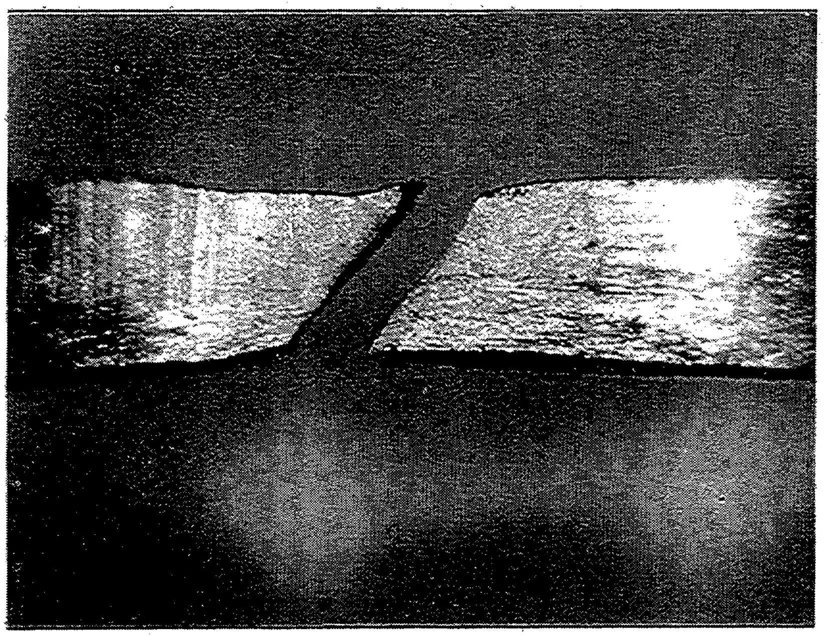
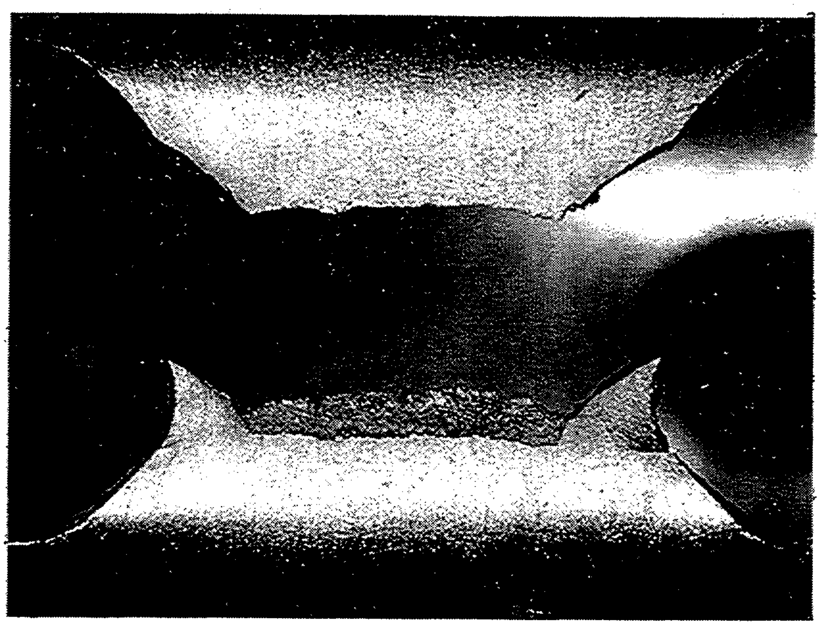


Figure 6: Cladding limit strain as a function of imperfection severity as predicted by theoretical analysis [reference 10] and measured experimentally at room temperature.



(a)



(b)

Figure 7: Macroscopic photographs of (a) uniaxial tension ring specimen failure and (b) plane-strain tension specimen failure for room temperature, quasi-static loading.

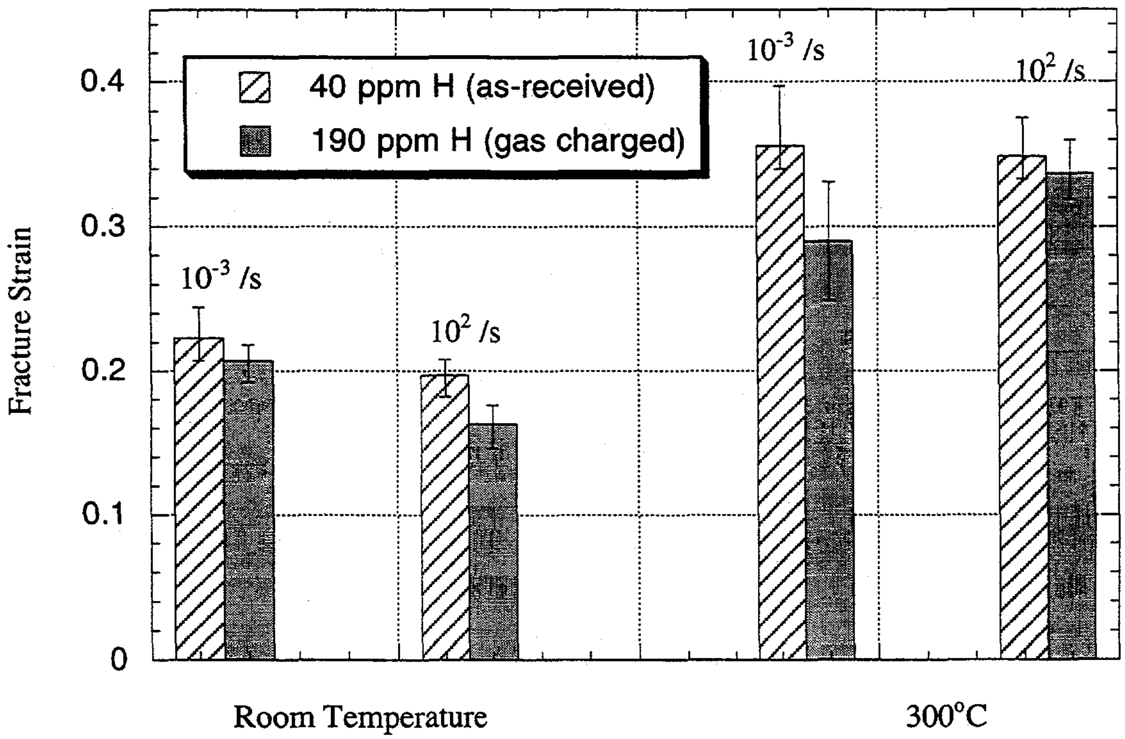


Figure 8: The effect of strain rate and 190 ppm hydrogen on the fracture strain of Zircaloy-4 cladding under transverse plane-strain loading.

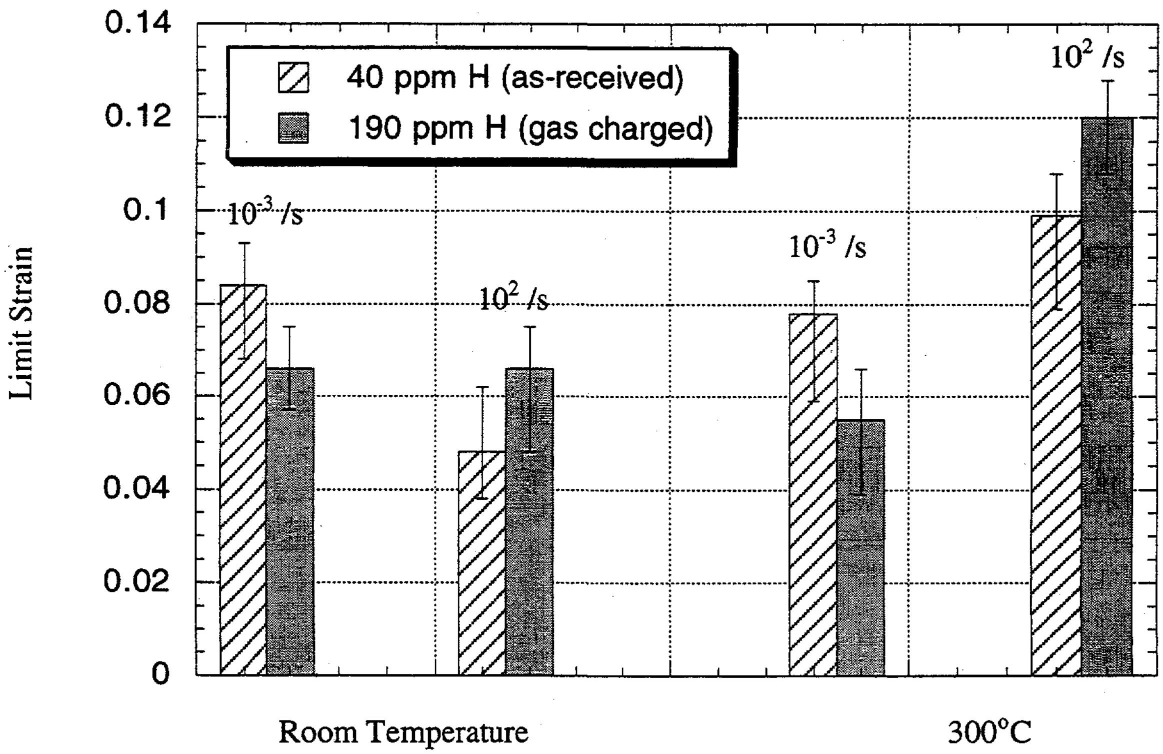


Figure 9: The effect of strain rate and 190 ppm hydrogen on the limit strain of Zircaloy-4 cladding under transverse plane-strain loading.

DEVELOPMENT AND VERIFICATION OF NRC's SINGLE-ROD FUEL PERFORMANCE CODES FRAPCON-3 AND FRAPTRAN

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ABSTRACT

The FRAPCON and FRAP-T code series, developed in the 1970s and early 1980s, are used by the U.S. Nuclear Regulatory Commission (NRC) to predict fuel performance during steady-state and transient power conditions, respectively. Both code series are now being updated by Pacific Northwest National Laboratory to improve their predictive capabilities at high burnup levels. The newest versions of the codes are called FRAPCON-3 and FRAPTRAN. The updates to fuel property and behavior models are focusing on providing best estimate predictions under steady-state and fast transient power conditions up to extended fuel burnups (> 55 GWd/MTU). Both codes will be assessed against a data base independent of the data base used for code benchmarking and an estimate of code predictive uncertainties will be made based on comparisons to the benchmark and independent data bases.

FRAPCON-3

The FRAPCON-3 code is an updated version of FRAPCON-2 that will be used by the NRC to audit vendor fuel performance codes with an emphasis on thermal, fission gas release, and rod internal pressure analyses. A code assessment of FRAPCON-3 has been recently concluded along with a peer review process that concentrated on those areas where the code will be applied for assessing licensing analyses, i.e., thermal and fission gas release. The code benchmarking data base includes thermal, fission gas release, internal rod void volumes, and cladding corrosion data that have previously been presented by Lanning, Beyer, and Painter (1997) and the soon to be published code integral assessment document (Lanning, Beyer, and Berna 1997). The code has also been assessed against an independent thermal and fission gas release data base that is provided in this paper. The differences in FRAPCON-3 and FRAPCON-2 predictions are illustrated by comparison to fission gas release data from experimental light water reactor fuel rods at moderate burnup levels.

The independent thermal data are divided into beginning-of-life (BOL) data (Table 1) and data as a function of nominal to high burnup levels (Table 2). The independent fission gas release data are summarized in Table 3. Presenting predicted temperatures minus measured temperatures as a function of linear heat generation rate (LHGR) at BOL for both the benchmarking and independent data bases (Figure 1) demonstrates that there is no bias in predictions. There is a slightly larger scatter in the comparison to independent data than to the benchmark data but the scatter is relatively small with a standard deviation of 24.5°C for the helium-filled rods increasing to 31.4°C when xenon-filled rods are included.

Table 1. Independent Data for BOL Fuel Temperatures (all BWR-size rods in Halden Reactor, Wiesenack 1996)

Gap Size, μm (and Gap-to-Diameter Ratio, %)	Initial Fill Gas Type and Room-Temperature Pressure, psia (MPa)	Maximum Rod-Average LHGR, kW/ft (kWm)
50 (0.47)	He 14.7 (0.10)	9 (30)
100 (0.94)	He 14.7 (0.10)	9 (30)
200 (1.9)	He 14.7 (0.10)	9 (30)
50 (0.47)	Xe 14.7 (0.10)	9 (30)
100 (0.94)	Xe 14.7 (0.10)	9 (30)
200 (1.9)	Xe 14.7 (0.10)	9 (30)

Table 2. Independent Data for Fuel Temperatures at Nominal-to-High Burnup

Reactor and Type (and Reactor for Ramping), Reference	Rod-Average Burnup, GWd/MTU	Rod Identification	Diametral Gap Size, μm (Gap-to-Diameter ratio, %)	Initial Fill Gas Type and Room-Temperature Pressure, psi (MPa)	Maximum Rod-Average LHGR kW/ft (kW/m)
Ringhals BWR (Halden), [a]	67	Halden, Rod 2	265 (2.5)	He 73 (0.50)	7.6 (25)
Quad Cities BWR (DR-2), Knudsen et al. 1993	42, 24	GE-2, GE-4 from RISØ-III	225 (2.1)	He 73 (0.5)	12.6 (41), 13.2 (43)
Halden (Halden), Chantoin et al. 1994	39	FUMEX-4A	220 (2.1)	He 44 (0.30)	15.7 (52)

[a] Halden Reactor Project. 1997. Personal communication with USNRC.

Table 3. Independent Data for FGR at Nominal to High Burnup

Reactor and Type (and Reactor for Ramping), Reference	Rod-Average Burnup, GWd/MTU	Rod ID	Diametral Gap Size, μm (Gap-to-Diameter Ratio, %)	Initial Fill Gas Type and Room-Temperature Pressure, psia (MPa)	Maximum (Bump Terminal) LHGR, kW/ft (kW/m)	Hold Time, hours	Measured FGR, %
Quad Cities BWR (DR-2), Knudsen et al.1993	43	GE-2 from RISØ-III	225 (2.1)	He, 97 (0.66)	12.6 (40.7)	41	24.6
Quad Cities (DR-2), Knudsen et al. 1993	22	GE-4 from RISØ-III	225 (2.1)	He, 97 (0.66)	13.2 (43.3)	34	27.0
Quad Cities (DR-2), Knudsen et al. 1993	42	GE-6 from RISØ-III	225 (2.1)	He, 97 (0.66)	11.6 (37.9)	140	26.0
Halden (Halden), Chantoin et al. 1994	48.8	FUMEX Case 6s	260	He, 370 (2.5)	~15 (50)	83 days	50
Halden (Halden), Chantoin et al. 1994	48.8	FUMEX Case 6f	260	He, 370 (2.5)	~12 (40)	150 days	45
ANO-1 (Studsvik), Wesley et al. 1994	62.3	R1	188 (2.0)	He, 400 (2.7)	12.0 (39.5)	12	9.3
ANO-1 (Studsvik), Wesley et al. 1994	62.3	R3	188 (2.0)	He, 400 (2.7)	12.9 (44)	12	11.2

Predicted temperatures minus measured temperatures at the fuel centerline as a function of burnup are provided in Figure 2 for both the benchmark and independent data. Because there are large variations in the LHGRs of the fuel rod data associated with this figure, this makes it difficult to assess the relative degree of under/overprediction. Therefore, the ratio of the difference between predicted and measured temperatures to the difference between measured centerline and coolant temperatures as a function of rod-average burnup is

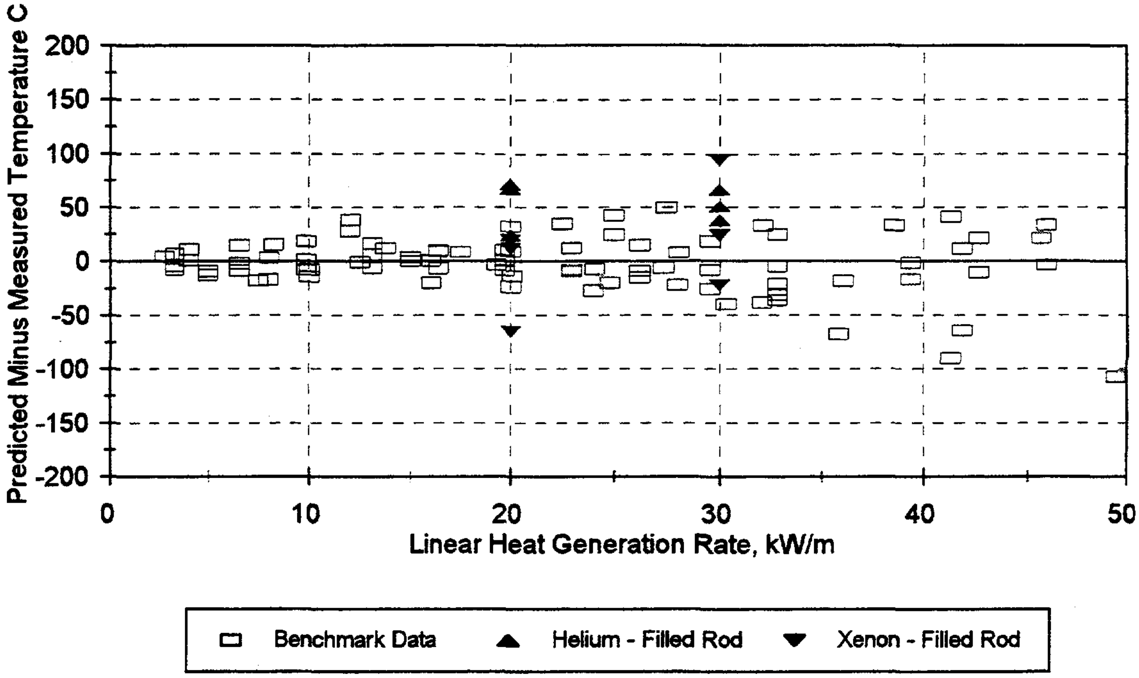

Figure 1. FRAPCON-3 Predicted-Minus-Measured Centerline Fuel Temperature at BOL as a Function of LHGR for Both Independent and Benchmark Data Sets

provided in Figure 3. As demonstrated by Figures 2 and 3, FRAPCON-3 significantly underpredicts (17 to 25%) the two RISØ-III rods at rod-average burnups of 22 and 42 GWd/MTU. The reason for the underprediction is unknown and does not agree with the relatively good prediction of the FUMEX rod from the independent data at 39 GWD/MTU and the benchmark data at burnups below 45 GWd/MTU. The independent data from the Halden ramped rod at a rod-average burnup of 67 GWd/MTU is underpredicted by 7.5 to 17.5% which is consistent with the one rod from the benchmark data (Halden Ultra-High-Burnup rod) at burnups >45 GWd/MTU.

The code fission gas release comparisons to both the benchmark and independent fission gas release data are provided in Figures 4 and 5. The independent data are predicted to within 5% release (absolute) of the measured values, except the RISØ rod at 22 GWD/MTU burnup, which is better than the power ramped rods in the benchmark data base. The relatively good prediction of the RISØ rod at 42 GWd/MTU contrasts with the fact that centerline temperatures for this rod were significantly underpredicted (>200°C) at the ramp terminal powers. The standard deviation on fission gas release is 5.4% for both benchmark and independent data, if rods with unstable (densification prone, >2.5% TD) fuel and BWR commercial rods (with large uncertainty in rod powers) are eliminated from the benchmark data.

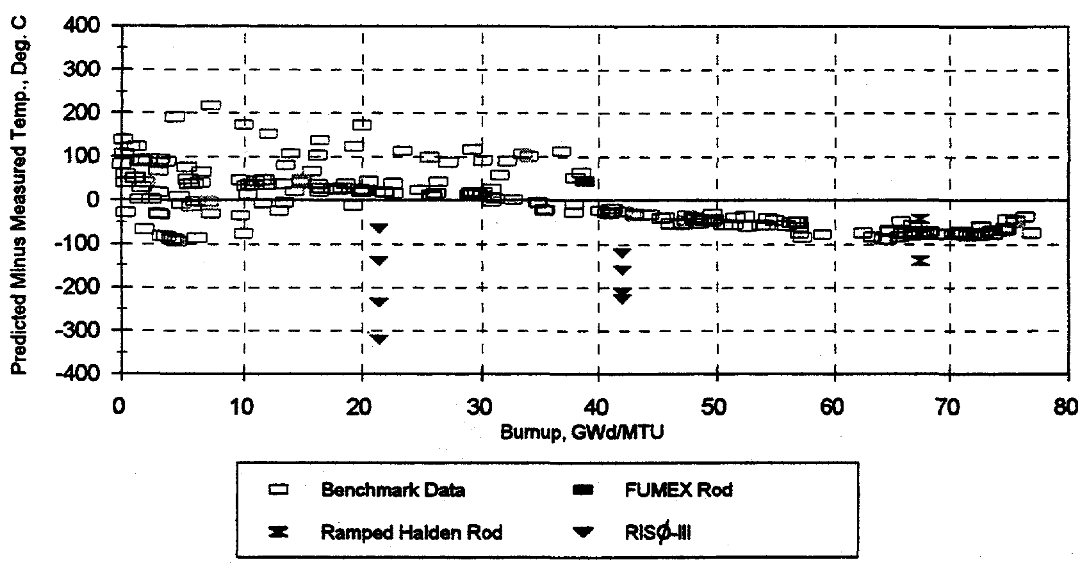

Figure 2. Predicted-Minus-Measured Fuel Center Temperatures as a Function of Burnup for Benchmark Data and Independent Data Sets

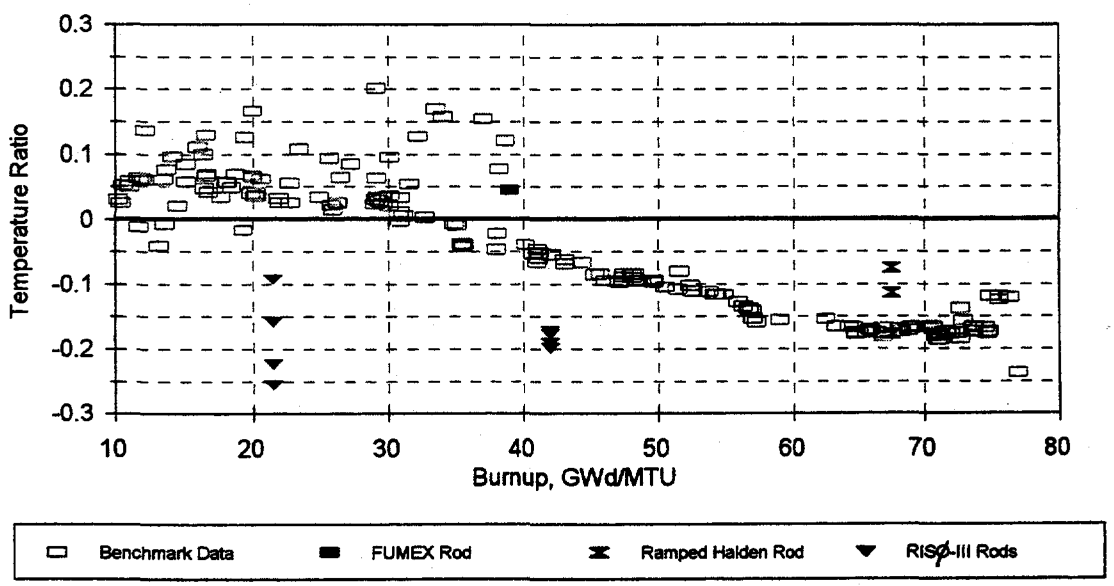

Figure 3. Ratio of FRAPCON-3 Predicted-Minus Measured Divided by Measured Fuel Centerline-Minus-Coolant Temperatures as a Function of Fuel Burnup for Benchmark Data and Independent Data Sets

A comparison is also provided to demonstrate the predictive differences between the new updated FRAPCON-3 code and the FRAPCON-2 code. The difference between FRAPCON-3 and FRAPCON-2 predicted fission gas release is illustrated in Table 4 where predicted values for each code are compared to actual measured values of fission gas release from four rods with moderate burnup levels between 30 to 49 GWd/MTU. The latter two rods in Table 4 (IIIi5 and F14-6) are from the benchmark data while the first two rods (M2-2C and PA 29-4) are from Bagger, Carlson and Knudsen (1978). This table shows that FRAPCON-2 significantly underpredicts fission gas release while FRAPCON-3 predicts these rods relatively well. It is noted, though not shown here, that the FRAPCON-3 code predicts significantly higher fuel temperatures at high burnups (>45 GWd/MTU) than FRAPCON-2.

In summary, the FRAPCON-3 code provides a relatively good prediction of all the thermal data below a burnup of 45 GWd/MTU except the RISØ-III rods and begins to underpredict the two rods from the benchmark and independent data above 45 GWd/MTU. The code provides a very good prediction of fission gas release for fuel rods with stable fuel (low fuel densification, <1.5% TD) and accurate estimates of rod power. FRAPCON-3 provides a much better prediction of fission gas release at moderate to high burnup levels than FRAPCON-2 and better thermal predictions at high burnup due to the inclusion of the effects of fuel thermal conductivity degradation. The code documentation including the model description document, code manual with input instructions, and code integral assessment (Lanning, Beyer, and Painter 1997; Berna et al. 1997; and Lanning, Beyer, and Berna 1997) are soon to be published by the NRC.

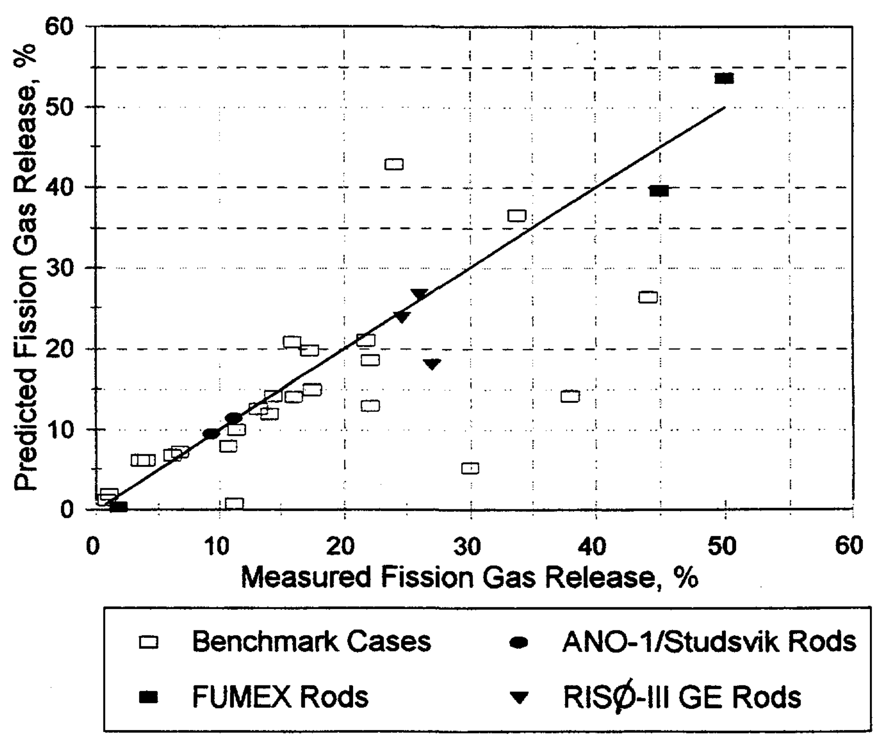

Figure 4. Predicted versus Measured Fission Gas Release for Benchmark Steady-State/Power-Ramp Data and Independent Data Sets

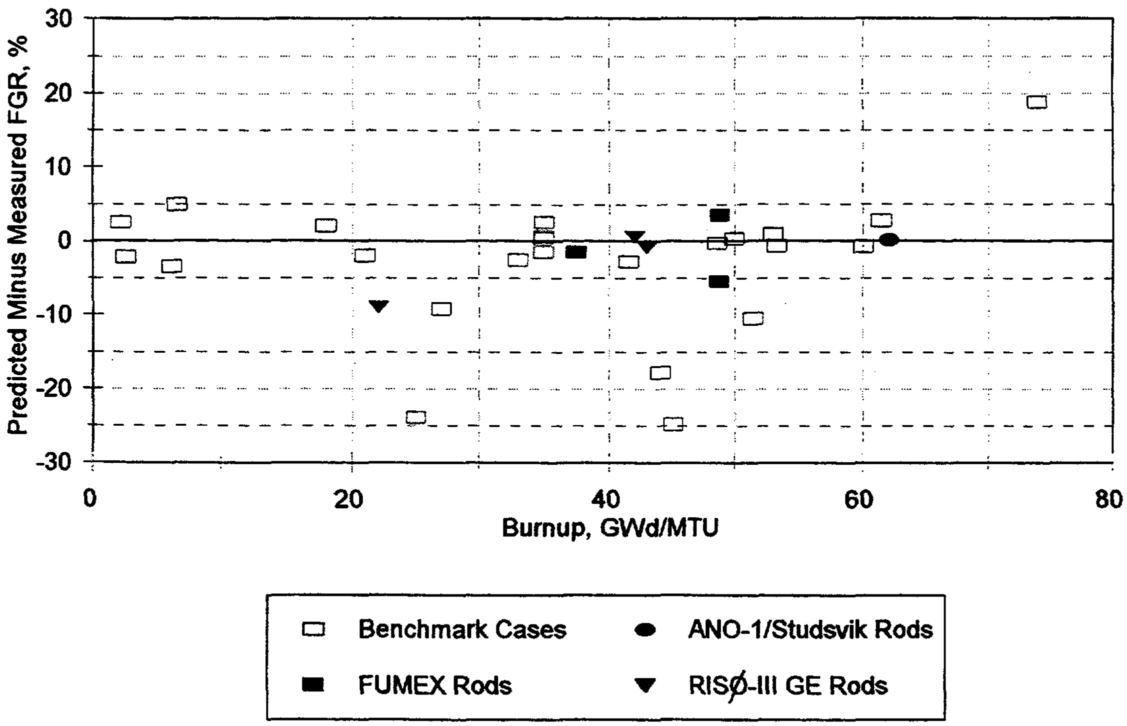

Figure 5. Predicted-Minus-Measured FGR as a Function of Burnup for Benchmark Steady-State/Power-Ramp Data and Independent Data Sets

Table 4. Comparison of Fission Gas Release Predictions from FRAPCON-2 and FRAPCON-3 with FRACAS-1 and "MASSIH"

Rod Number	Measured FGR, %	FRAPCON-2 Predicted FGR[1]	FRAPCON-3 Predicted FGR[2], %	Burnup GWd/MTU	FRAPCON-2 Predicted Minus Measured FGR, %	FRAPCON-3 Predicted Minus Measured FGR, %
M2-2C	35.6	23.5	36.5	43	-12.1	0.9
PA29-4	48.1	28.5	43.6	40.9	-19.6	-4.5
111i5	14.4	3.1	14.2	48.6	-11.3	-0.2
F14-6	22.1	0.16	12.7	30	-21.94	-9.4

(1) Using PARAGRASS fission gas release and FRACAS-2 models

(2) Using Massih fission gas release and FRACAS-1 models

FRAPTRAN

FRAPTRAN is an updated version of the FRAP-T (Fuel Rod Analysis Program-Transient) code series developed by the Idaho National Engineering Laboratory (INEL)[1] for the NRC to calculate the transient fuel performance of single fuel rods. As with FRAPCON-3, the objective of the work is to implement improved burnup-dependent thermal and mechanical models and correct other recognized deficiencies. FRAPTRAN is being developed from FRAP-T6, the sixth version in the FRAP-T code series, that was released in May 1981 (Siefkin et al. 1981) with an update in 1983 (Siefkin et al. 1983). Since that time there have been few modifications, with the result that FRAP-T6 has not incorporated changes to accommodate high burnup fuel behavior which have been incorporated in other codes.

Work on FRAPTRAN began by PNNL in FY-1997. The updates are to be parallel and consistent with the changes that have been done to FRAPCON-3 (Lanning, Beyer, and Painter 1997). As already noted, the emphasis of the work is on improving the high burnup predictive capability of the code. Other requirements placed on the FRAPTRAN development work include: retaining the capability to model both pressurized-water and boiling-water reactor conditions; retaining the capability to model the thermal effects of mixed oxide fuel; being capable of modeling a wide range of transients such as reactivity initiated accidents, loss of coolant accidents, and anticipated transient without scram;continuing the coding in FORTRAN; and making the code as independent of computer platform as possible.

Among the issues and drivers for developing FRAPTRAN are that general updates were needed to account for the effect of high burnup on properties, models, and fuel rod behavior; updates were needed to account for effects that are now understood to be important to fuel behavior during transients; and work was needed to be done for general coding improvement. Previous assessments of the FRAP-T6 code had observed that the code: overpredicted cladding hoop strain and ballooning strains; overpredicted fuel temperatures when a rod was filled with fission gas; overpredicted transient fission gas release using the PARAGRASS model; and that the FRACAS-II mechanical subcode needed work.

Modifications to the code are being grouped into three general areas. First, general coding improvements to address known errors, ensure consistency across the code, and to delete undesirable or no longer needed coding/models. Second, model updates to existing models to account for data, knowledge, etc., gained since FRAP-T6 was released (e.g., radial power distribution, contact conductance, and burnup dependent material properties like fuel thermal conductivity). And third, new model additions to extend the applicability of FRAPTRAN (e.g., modeling to account for fission gas release during fast transients).

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(1) INEL is now known as the Idaho National Engineering and Environmental Laboratory (INEEL).

The FRAPTRAN modifications began with FRAP-T6, Version 21. This code (plus Version 22) was transferred to PNNL and installed in September 1996. The FRAPTRAN modifications accomplished during FY-1997 have included the following:

- An initialization link has been established between FRAPCON-3 and FRAPTRAN. This link consists of a file written by FRAPCON-3 that contains selected burnup dependent values such as radial power and burnup profiles for each axial node, gas composition and pressure, permanent cladding and fuel strains, and other variables. This file is read by FRAPTRAN to initialize the burnup dependent variables. To be consistent with the ability to read this data from the FRAPCON-3 file, gadolinia concentration for the fuel and axially varying radial power and burnup profiles can now also be entered through the input data file.

- A revised fuel thermal conductivity model has been implemented using the same model used in FRAPCON-3 (Lanning, Beyer, and Painter 1997). This model has both gadolinia and local burnup dependencies. Because of the local burnup dependence of the new model, the fuel thermal conductivity model is called every time a value is needed rather than interpolating a fuel thermal conductivity table that is temperature dependent only.

- The change in contact gap conductance implemented in FRAPCON-3 (Lanning, Beyer, and Painter 1997) has been implemented in FRAPTRAN. The balance of the gap conductance modeling was reviewed to assure consistency.

- The MATPRO-11 (Hagrman, Reymann, and Mason 1981) versions of gas thermal conductivity and gas viscosity have been implemented to replace the MATPRO-9 versions used in FRAP-T6. This maintains the compatibility with FRAPCON-3.

- Selected coding options have been deleted from FRAPTRAN. These include the uncertainty sensitivity analysis, the failure mode analysis (FRAIL subcode package), and the licensing assistance evaluation model package.

- Values of constants, such as pi and a factor to convert thermal conductivity from British units to SI units, have been standardized. Values in FRAP-T6 were found to be inconsistently defined with four or more significant digits, or even values that varied by a few percent.

Work still needing to be done, and planned for FY-1998, includes: incorporating burnup-dependent mechanical properties and models; reviewing and revising as necessary the cladding oxidation models; developing and implementing a transient fission gas release model to replace PARAGRASS; and other general improvements to the coding. Preparing draft documentation and a draft code assessment is also planned for FY-1998.

Test cases and experimental data are being collected for the future verification and validation of FRAPTRAN. Cases that are being reviewed and proposed for this effort include: cases used in the assessment of FRAP-T6 (Chambers et al. 1981), the large-break

LOCA tests run in the NRU reactor (MT-1, MT-3, MT-4, and/or MT-6A), selected tests conducted in the Halden Boiling Water Reactor (e.g., IFA-508), selected tests conducted in the Power Burst Facility (e.g., RIA tests, LB-LOCA tests, and operational transient tests), pellet-cladding interaction ramp tests conducted by the Fuel Performance Improvement Program, recent RIA experimental tests such as those conducted at the Cabri and NSRR facilities (ANS 1997), and others yet to be identified. Measured parameters that will be of interest include measured fuel and cladding temperatures, cladding strains, fill gas pressure, and fuel rod rupture times.

Although formal assessment of FRAPTRAN has not yet begun, some preliminary comparisons of FRAPTRAN against data have been conducted. One example is the initial power ascension of Rod 1 from IFA-432 irradiated in the Halden Boiling Water Reactor. Rod 1 was a simulation of a standard boiling water reactor rod. Presented in Figure 6 is a comparison of measured fuel centerline temperature as a function of linear heat generation rate against temperatures calculated by FRAPCON-3, FRAP-T6, and FRAPTRAN. It may be seen that the FRAPTRAN calculated temperature is in fair agreement with the measured temperature and the temperature calculated by FRAPCON-3. A second example is the effect of initializing FRAPTRAN from a FRAPCON-3 run prior to calculating a power ascension. Presented in Figure 7 are fuel radial temperature profiles at 20 kW/m based on different initialization burnup levels as calculated by FRAPCON-3. The effects of decreased thermal conductivity and highly peaked radial power profile are well demonstrated by comparing the zero and high burnup radial temperature profiles.

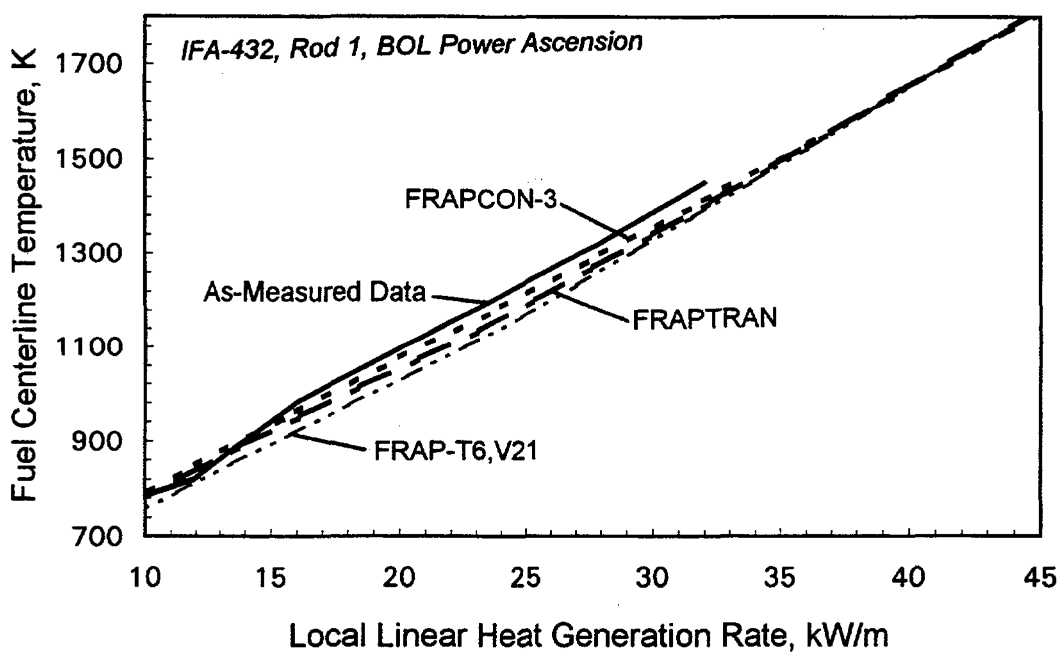

Figure 6. Comparison of Measured and Predicted Fuel Centerline Temperature for Initial Power Ascension of Rod 1, IFA-432

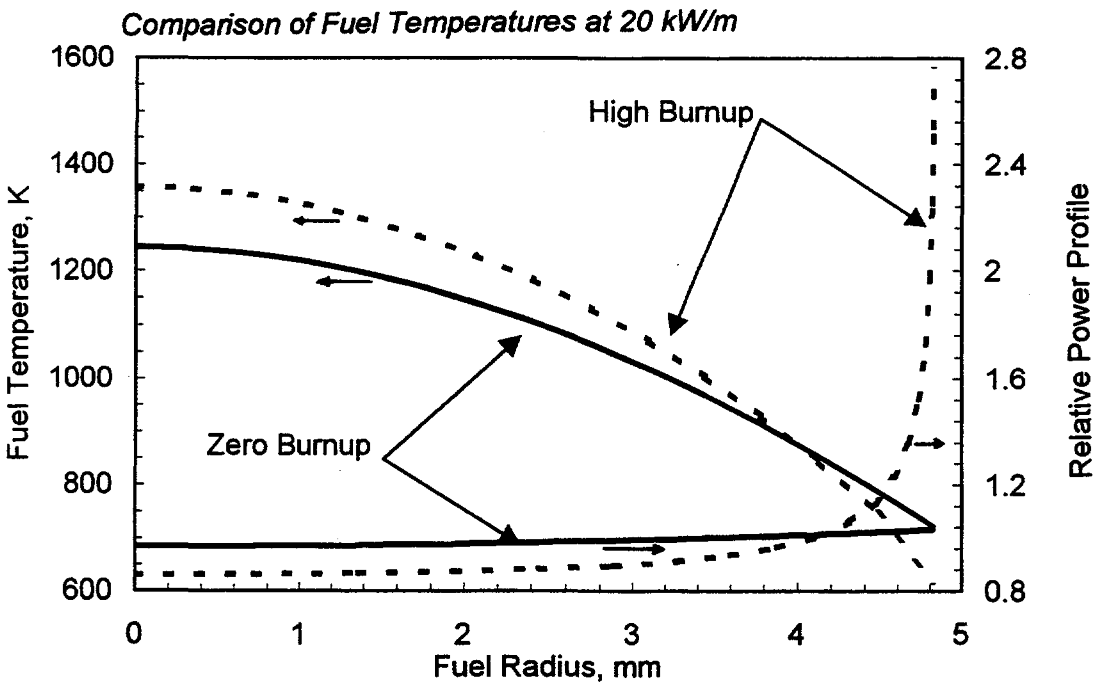

Figure 7. Comparison of Beginning-of-Life and End-of-Life Radial Temperature Profiles

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The Status of the RIA Test Program in the NSRR

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To provide a data base for the regulatory guide of light water reactors, behavior of reactor fuels during reactivity-initiated accident (RIA) conditions is being studied in the Nuclear Safety Research Reactor (NSRR) program of the Japan Atomic Energy Research Institute (JAERI). Recent results obtained from the NSRR experiments with irradiated PWR and BWR fuels are described and discussed in this paper. The results from the most recent experiments with high burnup PWR fuels with low-tin cladding resulted in significant fuel deformation, cladding failure and mechanical energy generation.

I. INTRODUCTION

Experimental program at the NSRR(1-3) and at the CABRI(4,5) appear to indicate that cladding failures may occur at enthalpy values lower than would be expected. Results from two experiments, i.e. HBO-1 and CABRI REP-Na1, raised concerns that existing licensing criteria for power-producing light water reactors could be inappropriate beyond a certain level of burnup.(6) This paper describes results from newly-conducted NSRR experiments including TK test series for 50 MWd/kgU PWR fuels with 1.3%Sn Zircaloy-4 cladding and FK test series for 45 to 55 MWd/kgU BWR fuels.

From year 1994 to 1996 we have performed seven pulse-irradiation experiments with high burnup PWR fuels as HBO test series(1,3), and observed fuel failures at low enthalpy level in two experiments HBO-1 and -5. Test conditions of the HBO experiments are listed in Table 1. The HBO-1 and -5 failed at 60 cal/g and at about 70 cal/g, respectively. Figure 1 shows horizontal cross-section of post-test failed cladding of the HBO-5. In the test fuel rod, significant hydride deposition below the oxide film generated in the cladding peripheral region. Brittle fracture can be seen in the cladding outer region where a number of hydride clusters precipitated, and ductile fracture appears in the inner region. SEM photographs in Fig. 1 also show brittle and ductile nature in the outer and inner fracture surfaces. Incipient cracking occurs in the outer, hydrided region, and propagates to the inner region. In addition to the through-wall crack, a number of micro-cracks perpendicular to the surface were found in oxide layer and heavily hydrided region. The through-wall crack could originate from one of these crack tips. The micro-cracks in oxide and hydride layers were observed also in the HBO-6 and HBO-7 which resulted in no failure. Figure 2 shows the micro-cracks observed in the HBO-5 and -6. Figure 3 illustrates the relative elevation where the test fuel rod was sampled in each experiment, and occurrences of fuel failure and micro-crack generation. Fuel failure occurred in the experiments with test fuel rods from the highest elevation, and micro-crack generation in oxide and hydride layers observed in the experiments with rod from the relatively high level. The micro-cracks observed in the HBO-6 and -7 are expected to be a precursor of cladding failure. It can be naturally accepted that occurrence of fuel failure in the HBO experiments correlates with the sampling elevation, accordingly, with thickness of oxide film and severity of hydrogen deposition of the tested fuel rod.

II. TK TEST SERIES (38 to 50 MWd/kgU PWR fuels with 1.3%Sn Zry-4 cladding)

The fuel rods in the HBO series were sampled from a mother rod irradiated in 48 MWd/kgU lead-use program, and these rods had 1.5%Sn Zircaloy-4 cladding. However, fuels with 1.3%Sn (low tin) Zircaloy-4 cladding were adopted for 48 MWd/kgU regular-use in Japanese PWRs. Accordingly, we started new series of experiments with the fuels with the low tin cladding, as TK test series. Test conditions of first four TK experiments are listed in Table 1.

II.1 TK-1

The test fuel rod of the TK-1 was sampled from 5th span (from the top) of 17x17 PWR type-A fuel ('type-A' denotes fuel manufactured by Mitsubishi Heavy Industries, Ltd.) with 1.3%Sn Zircaloy-4 cladding. The mother rod had been irradiated for two cycles in Takahama unit 3 reactor of Kansai Electric Power Company, Inc. Burnup of the test fuel is 37.8 MWd/kgU. Because of the relatively low burnup and low sampling elevation, oxide thickness of the cladding remains 7 μm. Fuel enthalpy during the pulse-irradiation reached 125 cal/g at maximum, and the rod did not fail. Figure 4 shows transient histories of the cladding surface temperature during the transient. Cladding surface temperature increased rapidly at the pulse, and DNB (departure from nucleate boiling) occurred. The temperature reached about 600 deg C at maximum. Figure 5 shows post-test appearance of the TK-1 test fuel rod. Significant swelling occurred over the fuel active region. The increase in cladding diameter is 10% in average over the pellet stack region, and 25% at maximum. X-ray photograph in Fig. 5 shows a portion where the most significant swelling appeared. As seen in the photograph, a gap between fuel pellets and cladding inner surface was not opened, and the fuel pellets swelled significantly. This fact suggests that the large increase in cladding diameter of the TK-1 rod is not caused by pressure increase in fuel rod plenum (ballooning), but produced by pellet swelling (PCMI: pellet/cladding mechanical interaction). Roughly polished radial and vertical cross-sections of the post-test TK-1 fuel are shown in Fig. 6. Large cracks and openings are observed in the cross-sections of the post-test fuel pellets. The results of the HBO tests indicated that rapid expansion of inter-granular fission gas caused grain-boundary separation, and it resulted in fuel pellet swelling and PCMI. On the other hand, the TK-1 suggests that prompt release of fission gas and subsequent increase of fuel pellet internal pressure contribute to the large deformation of the rod, possibly in combination with the early PCMI loading. Figure 7 shows residual hoop strain in NSRR/PWR fuel experiments as a function of peak fuel enthalpy. The higher fuel enthalpy correlates with the larger strain, and that of TK-1 is extremely large. Fission gas release in the TK-1 is about 20%. Figure 8 shows fission gas release as a function of peak fuel enthalpy. In the previous three HBO experiments HBO-2, -3 and -4 with type-A 50 MWd/kgU PWR fuels, fission gas release ranged from 17.7% to 22.7%. The fission gas release in the TK-1 is in this level, and the data point is located on an extension line of the data from MH, GK, OI, HBO-6 and -7 experiments.

II.2 TK-2 and -3

The recent experiments TK-2 and -3 were performed at only several weeks before the 25th WRSM, on October 1 and 8, 1997. The test fuel rods of the experiments are 17x17 PWR type-B fuels ('type-B' denotes fuel manufactured by Nuclear Fuel Industries, Ltd.) with 1.3%Sn Zircaloy-4 cladding. The mother rod of the TK-2 and -3 test fuels had been irradiated for three cycles also in Takahama unit 3 reactor, and burnup reached about 50 MWd/kgU. The test fuel rods of the TK-2 and -3 were sampled from 2nd span (from the top) and 4th span of the mother rod, respectively. Because of the difference in the sampling elevation, fuel burnup is 48 MWd/kgU for the TK-2 and 50 MWd/kgU for the TK-3. Oxide layer thickness of the cladding ranges from 15 to 35 μm for the TK-2 (23 μm in average), and from 4 to 12 μm for the TK-3 (7 μm in average). Both experiments were performed with $4.6 pulse in the NSRR, and fuel enthalpy reached about 99 cal/g in the TK-2 and 92 cal/g in the TK-3. As for the HBO-1 and -5, both resulting in fuel failure, possibility of influence from instrumentation (cladding elongation sensor in the HBO-1, cladding surface thermocouples in the HBO-5) on the cladding failures was discussed. To exclude the possibility, axial elongation sensors (both for cladding and fuel stack) and cladding surface thermocouples were not installed in the TK-2. Instead of the elongation sensors, float-type water column movement sensor was used to measure mechanical energy generation when the fuel rod failed during the experiment. As for the TK-3, the axial elongation sensors and cladding surface thermocouples were installed. The test conditions are identical between the TK-2 and TK-3 experiments except the instrumentation and the state of the test fuel rod.

The TK-2 experiment resulted in fuel failure. Figure 9 shows transient histories of the NSRR reactor power, signal from the water column movement sensor, fuel rod internal pressure and capsule internal pressure during the TK-2. During the pulse-irradiation, water column starts to move, and spikes appears in the both pressure histories simultaneously. These indicate that fuel failure occurred at that instant. When the fuel rod failed, fuel enthalpy reached about 60 cal/g. In the signal from the water column movement sensor, a half-wavelength corresponds to 3 mm displacement of the float at surface of coolant water. The thermal-to-mechanical energy conversion ratio, a ratio of mechanical energy generated to the peak fuel enthalpy of the fuel, is estimated as about 0.08%. A vertical crack over the fuel active region was observed in the post-test TK-2 fuel, as seen in Fig.10. The appearance of the crack is similar to that in the HBO-5. The X-ray photograph of Fig. 11 shows that most of fuel pellets remain inside the failed cladding, and it suggests that few fuel particle dispersed into the coolant water. The mechanical energy may be produced by gas released promptly from the fuel rod. This indicates that post-failure process in PCMI failure of high burnup PWR rod is similar to that in failure of water-logged fuel, i.e. low-temperature burst-type failure.

In the subsequent test TK-3, fuel failure did not occur. During the experiment cladding surface temperature reached about 700 deg C at maximum, as shown in Fig. 12. Although relatively large fuel deformation occurred, the cladding could survive in the experiment. Figure 13 shows residual hoop strain in NSRR/PWR fuel experiments as a function of peak fuel enthalpy (the TK-1 data is excluded in this figure, since the strain of the TK-1 is extremely large). The histories of cladding surface temperature in the TK-3 suggests that the fuel failure in the previous TK-2 occurred when cladding surface temperature remained low.

In the two experiments TK-2 and -3, only the test fuel rod sampled from the higher elevation, with thicker oxide layer and larger hydrogen pick-up, failed at about 60 cal/g. The results from the two experiments indicate that the critical factor is whether cladding has enough integrity, i.e. ductility, to survive by the time that cladding temperature reaches a certain level. In the experiments with test fuel rods sampled from the higher elevation, cracking occurs initially in radially-localized hydride layer of the cladding, and then propagates during early stage of the transient when cladding surface temperature remained low.

The type-A test fuel irradiated for three cycles is to be subjected to the next experiment TK-4. Figure 14 illustrates the relative sampling elevations of the TK test fuels.

III. FK TEST SERIES (41 to 56 MWd/kgU BWR fuels)

Only five experiments with 7x7 BWR fuels, i.e. TS test series[7], had been performed in the NSRR, and range of fuel burnup had been limited to 26 MWd/kgU. From FY1996, FK test series with 8x8 BWR fuels was started. The test fuels subjected to the FK-1 through FK-3 are from First Fukushima plant unit 3 reactor of Tokyo Electric Power Company, Inc.(TEPCO), and the test fuel of the FK-4 is from Second Fukushima plant unit 2 reactor of TEPCO. As listed in Table 2, fuel burnup ranges 41 to 45 MWd/kgU in the FK-1 through FK-3, and about 56 MWd/kgU in the FK-4. The FK-3 is to be performed on March 1998 and the FK-4 in FY1998. In already performed two experiments FK-1 and -2, fuel failure did not occur.

In the FK-1, fuel enthalpy reached 112 cal/g at maximum, and cladding surface temperature reached 360 deg C at maximum. Figure 15 shows transient histories of cladding surface temperature, fuel rod internal pressure and axial elongation of fuel stack and cladding. The axial elongation data show closure of gap between fuel pellet and cladding inner surface, and occurrence of PCMI. However, cladding diameter increase after the test in the FK-1 remained 0.85% in average, and 3% at maximum. Metallographies of round slices and vertical division from the FK-1 fuel are shown in Fig. 16. A number of radial cracks can be seen in the fuel pellet peripheral region. In the next test, FK-2, the peak fuel enthalpy is 60 cal/g. Thermocouple for the cladding surface temperature was not installed in the second test. The post-test FK-2 cladding does not have a diameter increase. Metallographies from the FK-2 fuel are shown in Fig. 17. Radial cracking in fuel pellet periphery is not significant in comparison with the FK-1, but a number of axial cracks are generated near the pellets center-line. Figure 18 shows residual hoop strain of post-test cladding as a function of peak fuel enthalpy. Because of larger gap between fuel pellet and cladding inner-surface in BWR fuels, cladding deformation in the BWR fuels are much smaller than those in the PWR fuels. Fission gas release during the FK-1 is about 8%, and about 3% in the FK-2. Figure 19 shows fission gas release as a function of peak fuel enthalpy. Large scattering of the BWR data could be due to difference in linear heat generating rate during the base-irradiation in the power-producing reactors.

IV. KEY QUESTIONS IN RIA FUEL BEHAVIOR AND ONGOING PROGRAMS

The results from the NSRR and CABRI REP-Na experiments have provided evidence that fuel failure of high burnup PWR fuel occurs at the low enthalpy level. The results indicate that decreased cladding integrity, fuel pellet deformation and increased internal pressure driven by fission gas expansion have significant role in the failure process. The NSRR experiments have also shown occurrence of post-failure events, including mechanical energy generation and fuel fragmentation.[2] To understand high burnup fuel behavior during RIA conditions, further efforts should be devoted to solve key questions including 1) mechanical properties of cladding materials, in particular, ductility reduction due to radially-localized hydride precipitation, 2) influence of pulse-width and initial temperature on the failure, 3) role of fission gas in fuel pellet swelling, grain boundary separation and fuel fragmentation, 4) assessment of post-failure events, 5) post-DNB failure modes and 6) code qualifications.

Hydrogen deposition, in particular, could have a critical importance in the high burnup PWR fuel behavior at an RIA. The effect of hydride deposition cannot be described by concentration averaged at cross-section, and the effect of radial and circumferential localization of the hydride clusters are very important. Photo-image analysis[8] for radial distribution of hydride clusters and hydrogen composition measurement on pre- and post-test cladding are being performed. Influence of radially localized hydride layer on the cladding ductility is being examined also in out-of-pile, separate-effect tests at JAERI. The separate-effect tests include highly-transient burst experiment and modified ring tensile test on machined specimen. In addition to as-fabricated sample, artificially hydrided cladding is being tested in the burst test and to be tested in the ring tensile test. As for the modified ring tensile test, optimum geometry of specimen and tooling is currently investigated.

Wide pulse-width and high coolant temperature can provide the higher cladding temperature when the cladding takes PCMI and/or high pressure loading. It is expected that reduction of hydride embrittlement occurs in the elevated cladding temperature, as is in accidental conditions of power-producing reactor. The NSRR does not have an answer regarding the pulse-width, but a high-temperature, high-pressure test capsule is in designing stage to clarify the initial temperature effect. Figure 20 shows a schematic of the high-temperature, high-pressure capsule for the NSRR experiment. Although the coolant is stagnant, the capsule produces high temperature and high pressure in PWR or BWR condition.

To investigate the role of fission gas in the failure process, measurements on radial distribution of accumulated fission gas and inter- and intra-granular inventory are continued with newly-installed PIE devices including ion micro analyzer. In addition to the examination regarding initial states of fission gas, fission gas release, in particular, possible prompt release should be studied. Out-of-pile fuel pellet heating test VEGA producing severe accident conditions by electrical heating will also provide information of fission gas state in high burnup fuels.

In the NSRR experiments, pressure and mechanical energy generation and fuel fragmentation have observed. To assess the post-failure events, JMH test series is continued. In the JMH tests, 20% enriched fuels had been irradiated in Japan Materials Testing Reactor (JMTR), and subjected to pulse-irradiation. In the recent test JMH-5, pressure pulse and mechanical energy were successfully measured. Several series of experiments with un-irradiated, fresh fuels are also being conducted in the NSRR. In these experiments, bare-pellet (without cladding) fuel and powder fuel are used to study the influence of contact modes between dispersed fuel and coolant water.

V. SUMMARY

The TK-1 experiment with 38 MWd/kgU PWR fuel with 1.3%Sn Zircaloy-4 cladding showed very large swelling, up to 25% increase in diameter, possibly caused by rapid expansion of fission gas accumulated in grain boundaries in combination with increase of fuel internal pressure due to prompt gas release. In the subsequent two experiments TK-2 and -3 with 48 to 50 MWd/kgU PWR fuels, only the test fuel rod sampled from the higher elevation, with thicker oxide layer and larger hydrogen pick-up, failed at about 60 cal/g. The results from the two experiments indicate that the critical factor is whether cladding has enough ductility to survive by the time that cladding temperature reaches a certain level. Although fuel dispersion was not significant, about 0.8% of energy generated was converted to mechanical energy in the TK-2.

The FK-1 and -2 experiments with 45 MWd/kgU BWR fuels resulted in no failure, and showed modest fuel deformation. A relatively large margin to PCMI loading can be expected in the BWR fuels since the larger gap between fuel pellet and cladding inner surface exists.

Figure 21 summarizes existing data from in-pile RIA experiments. In the NSRR program, pulse-irradiation experiments with high burnup PWR and BWR fuels are continued. To answer key questions in RIA fuel behavior, fresh fuel experiments and out-of-pile separate-effect tests also take important role.

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Table 1 Test conditions of tests HBO and TK (High burnup PWR fuels)

Test ID	Fuel Type	Span	Oxide Layer (μm)	Fuel Burnup (MWd/kgU)	Peak Enthalpy (cal/g)	Result
HBO-1	A	3rd	40 to 48	50.4	73	Failed at 60 cal/g
HBO-2	A	4th	30 to 40	50.4	37	No failure, FGR=17.7%
HBO-3	A	5th	20 to 25	50.4	74	No failure, FGR=22.7%
HBO-4	A	6th	15 to 20	50.4	50	No failure, FGR=21.1%
HBO-5	B	2nd	35 to 60	44	80	Failed at ~70 cal/g
HBO-6	B	4th	20 to 30	49	80	No failure, FGR=10.4%
HBO-7	B	3rd	N/A	49	80	No failure, FGR=8.5%
TK-1	A	5th	7	38	125	No failure, FGR=20%
TK-2	B	2nd	15 to 35	48	99	Failed at ~60 cal/g
TK-3	B	4th	4 to 12	50	92	No failure
TK-4	A	3rd	N/A	50	(100)	on February 1998

Span of 1st denotes the highest.

FGR is an acronym for fission gas release.

Values for HBO-5 through TK-4 are from preliminary evaluations.

Table 2 Test conditions of test FK (High burnup BWR fuels)

Test ID	Fuel Burnup (MWd/kgU)	Peak Enthalpy (cal/g)	Result
FK-1	45.4	112	No failure, FGR=8.2%
FK-2	45.4	60	No failure, FGR=3.1%
FK-3	41	(125)	To be performed on March 1998
FK-4	56	(90)	To be performed in the FY1998

Values are from preliminary evaluations.

FY1998 is a Japanese fiscal year from April 1998 to March 1999.

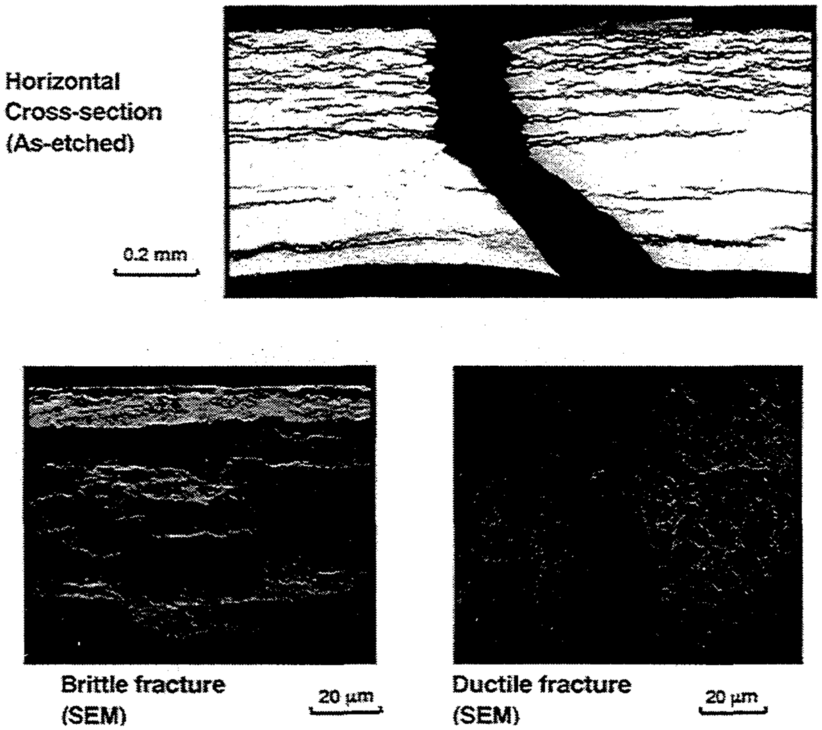

Fig.1 Through-wall crack observed in the HBO-5

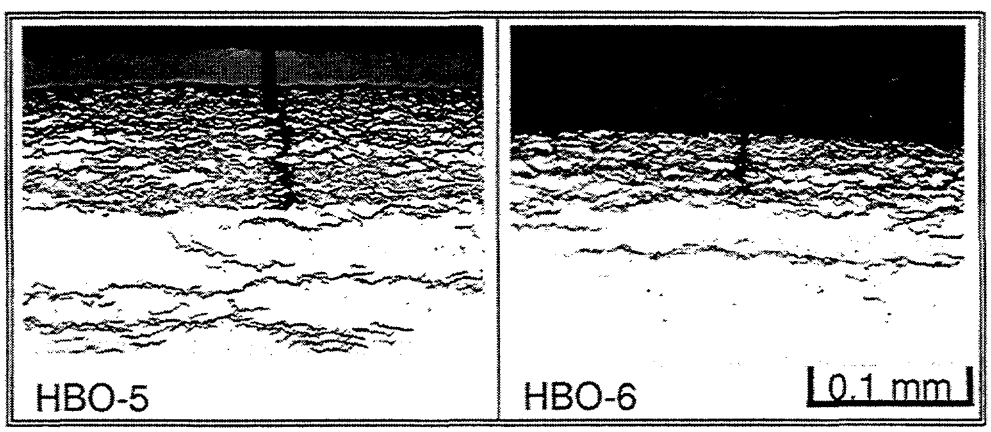

Fig.2 Micro-cracks observed in the HBO-5 and -6

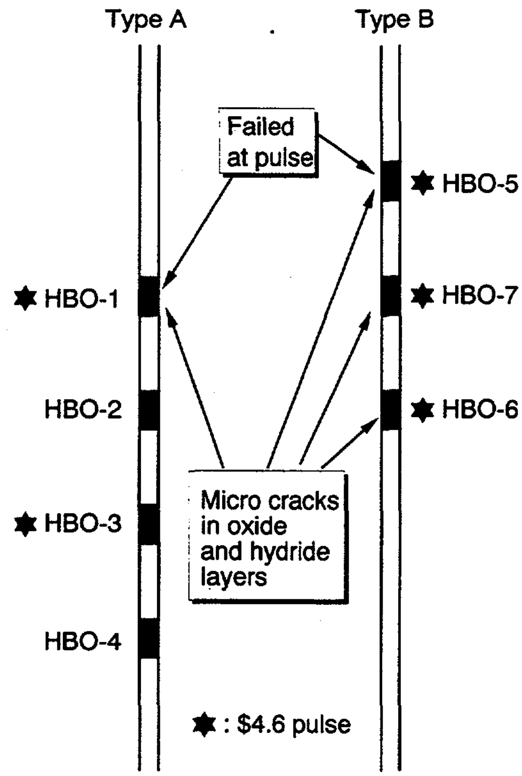

Fig.3 Sampling elevation of the HBO test fuels

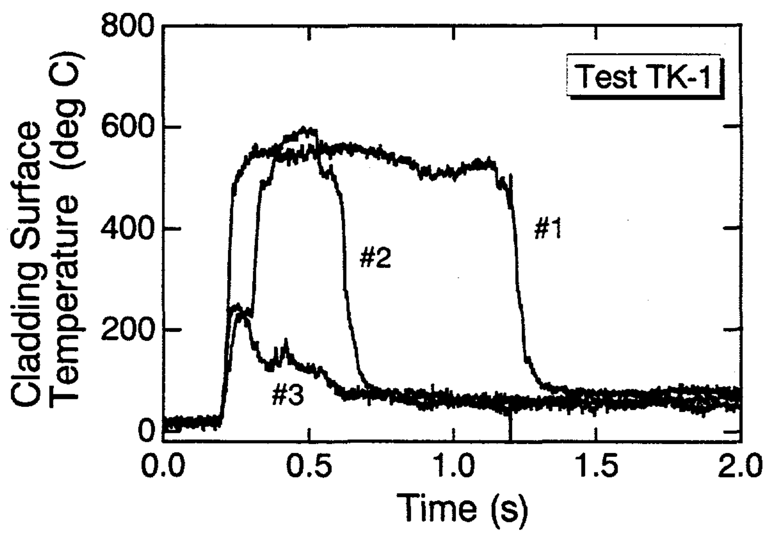

Fig. 4 Transient histories of cladding surface temperature in the TK-1

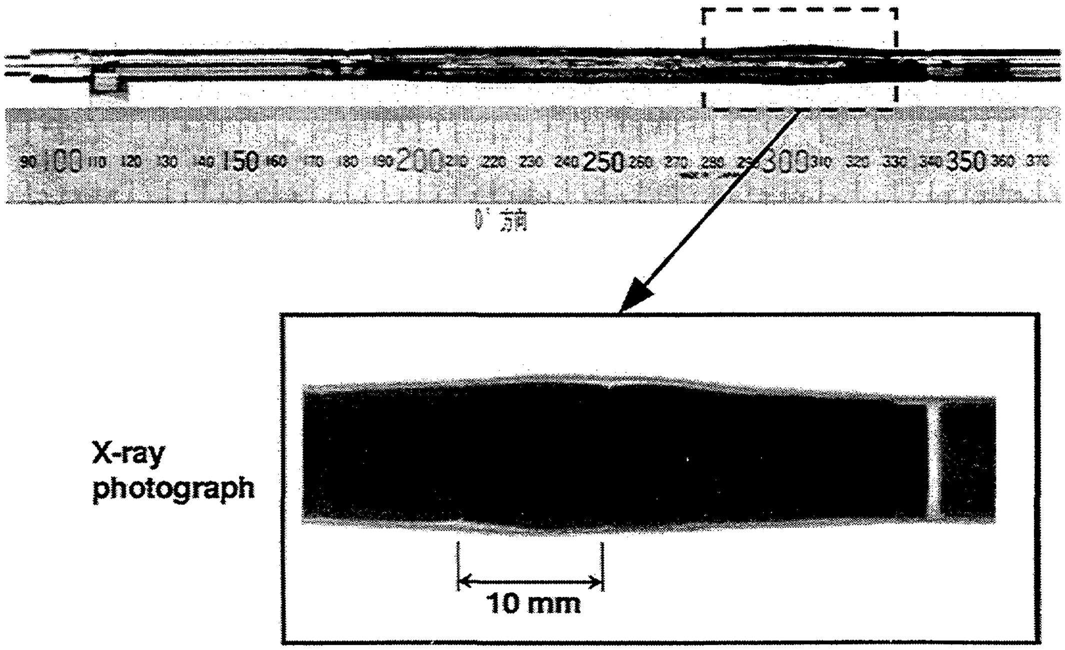

Fig. 5 Post-test appearance of the TK-1 test fuel

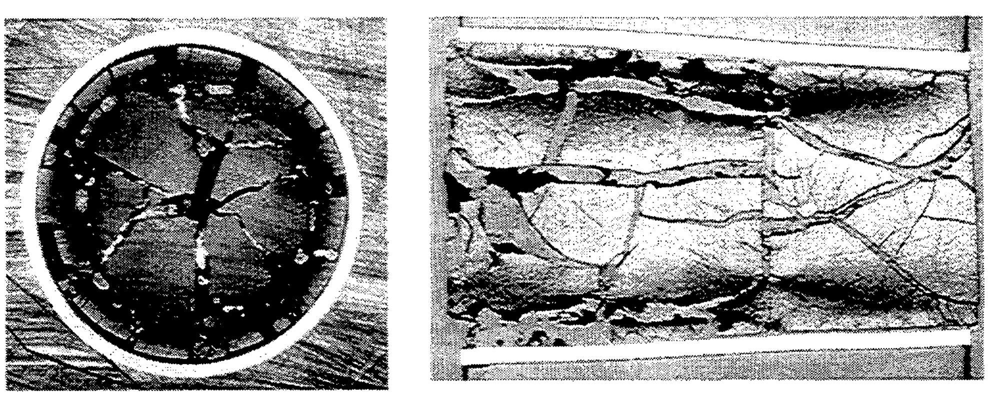

Fig. 6 Roughly polished cross-sections of the post-test TK-1 fuel

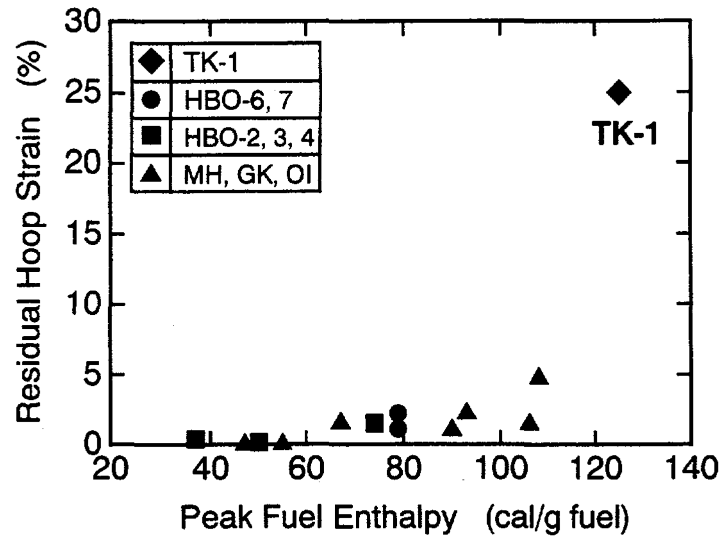

Fig. 7 Residual hoop strain in PWR fuel experiments as a function of peak fuel enthalpy

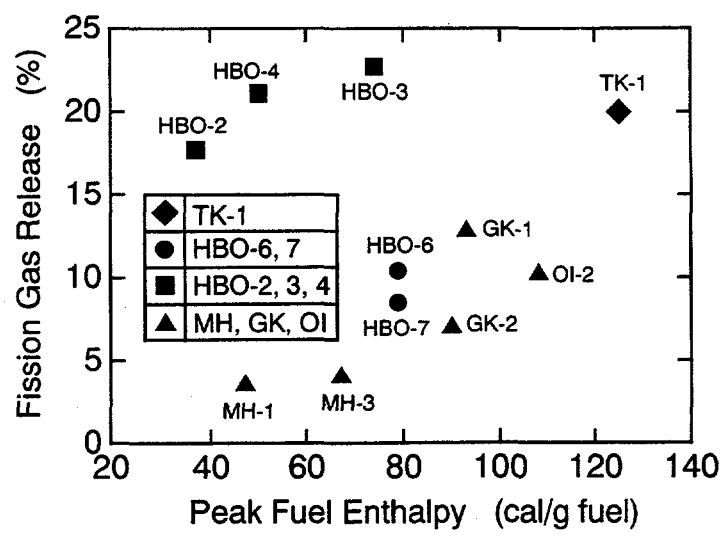

Fig. 8 Fission gas release in PWR fuel experiments as a function of peak fuel enthalpy

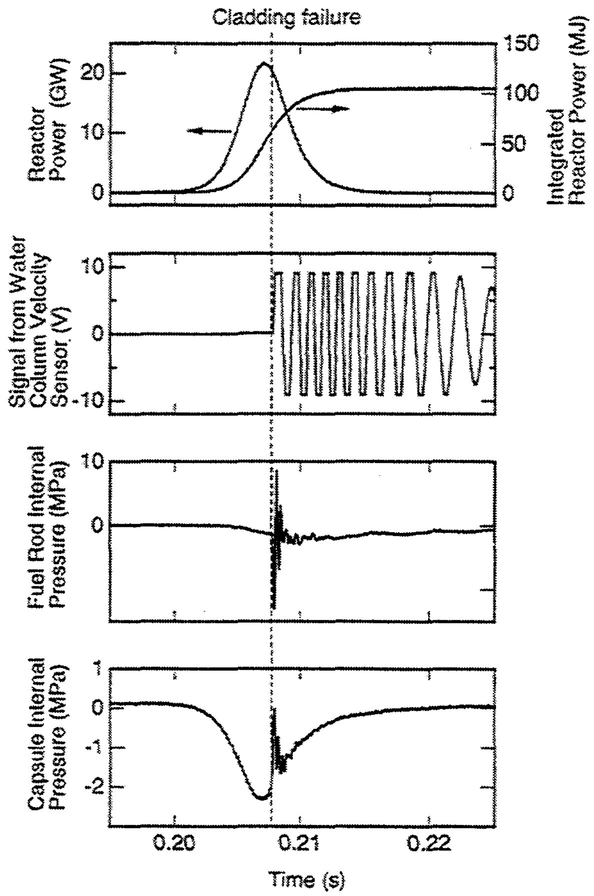

Fig. 9 Transient data of the TK-2 resulted in fuel failure

Fig. 10 Post-test appearance of the TK-2 test fuel

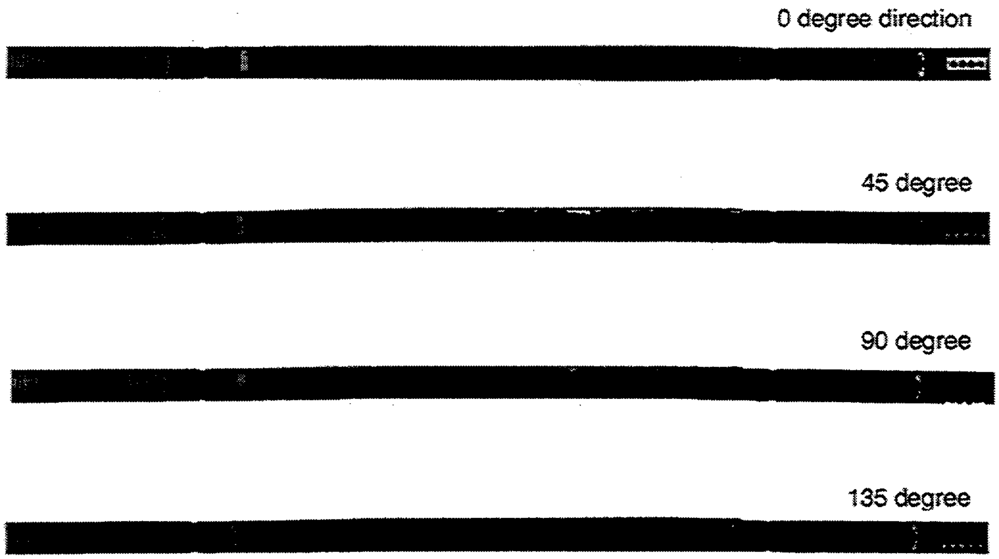

Fig. 11 X-ray photograph of the post-test TK-2 test fuel

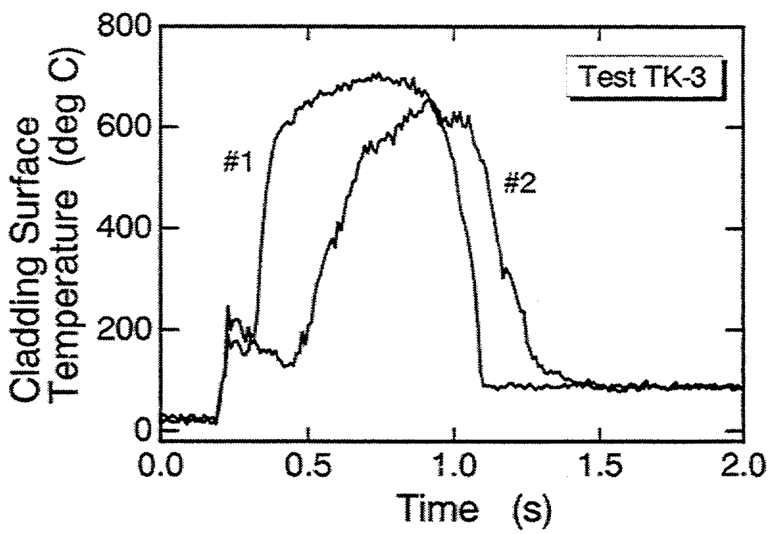

Fig. 12 Transient histories of cladding surface temperature in the TK-3

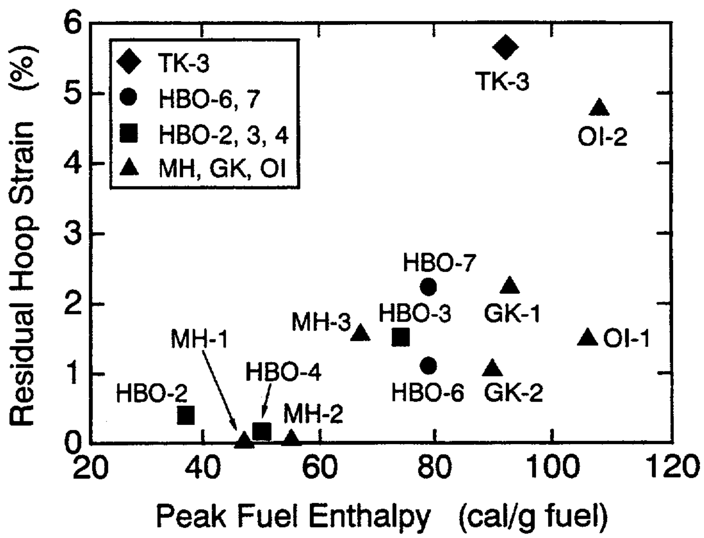

Fig. 13 Residual hoop strain in PWR fuel experiments (except TK-1) as a function of peak fuel enthalpy

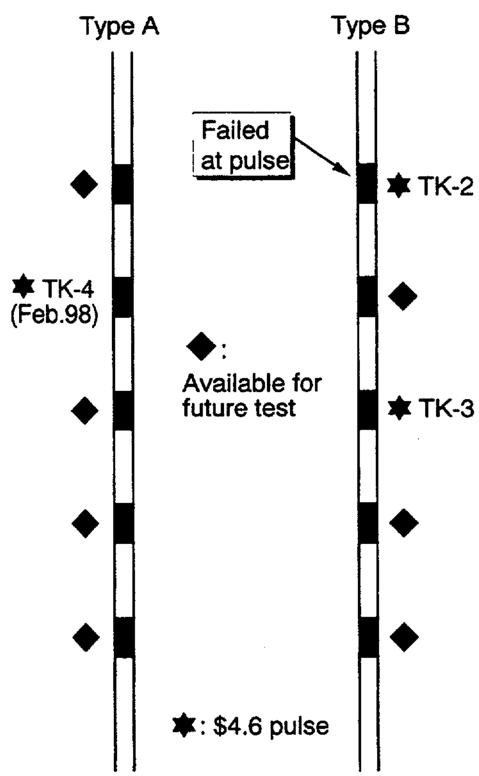

Fig. 14 Sampling elevation of the TK test fuels

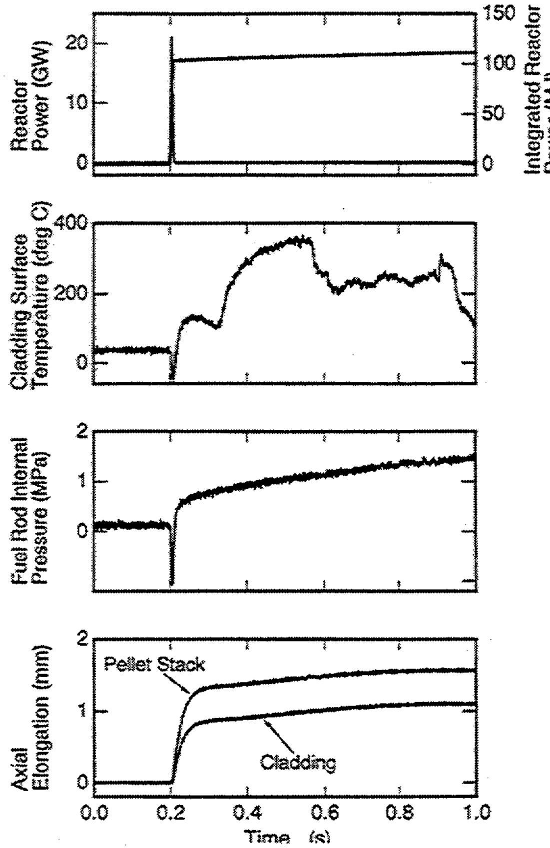

Fig. 15 Transient data of the TK-2 resulted in fuel failure

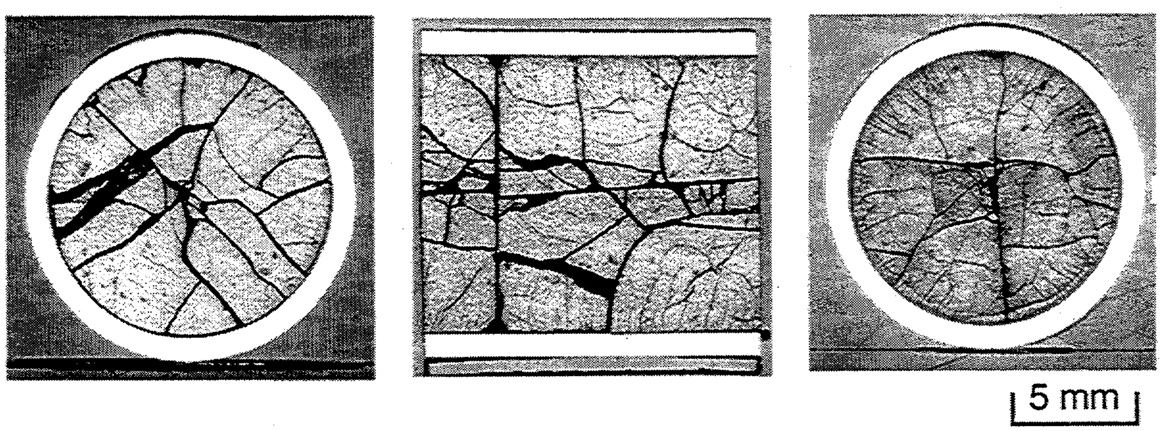

Fig. 16 Cross-sections of the post-test FK-1 fuel (as-polished)

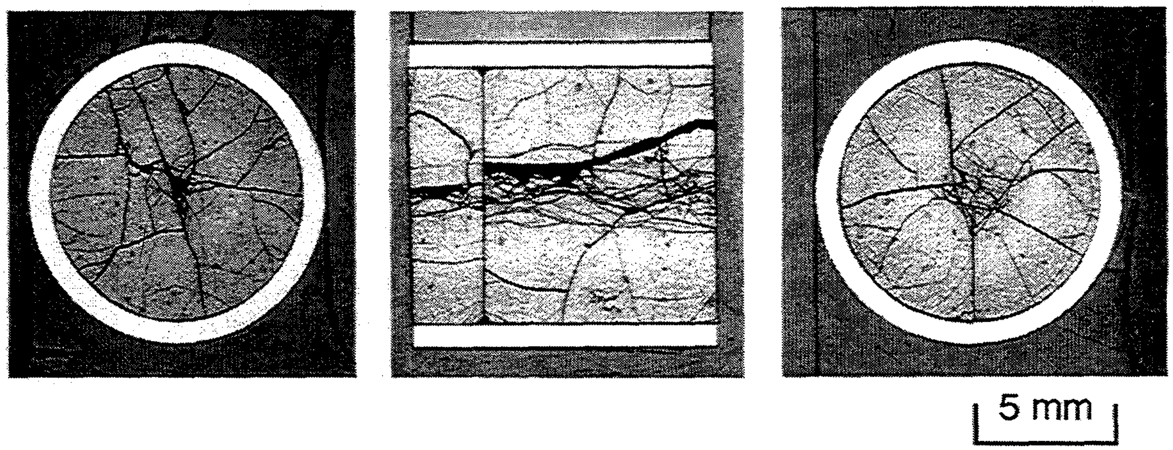

Fig. 17 Cross-sections of the post-test FK-2 fuel (as-polished)

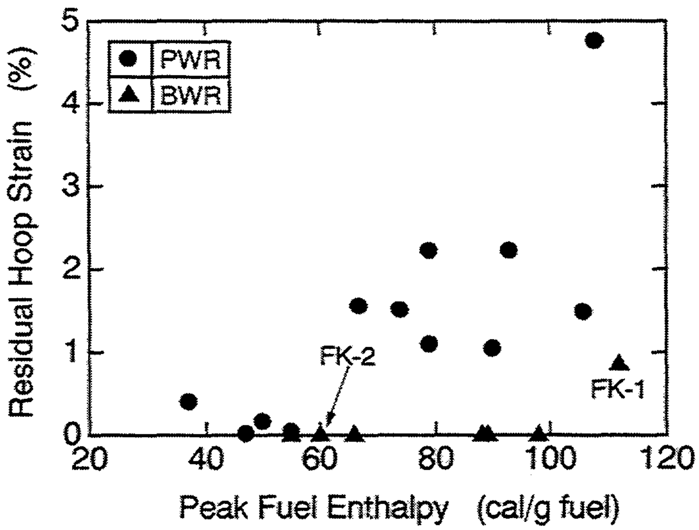

Fig. 18 Residual hoop strain in PWR and BWR fuel experiments

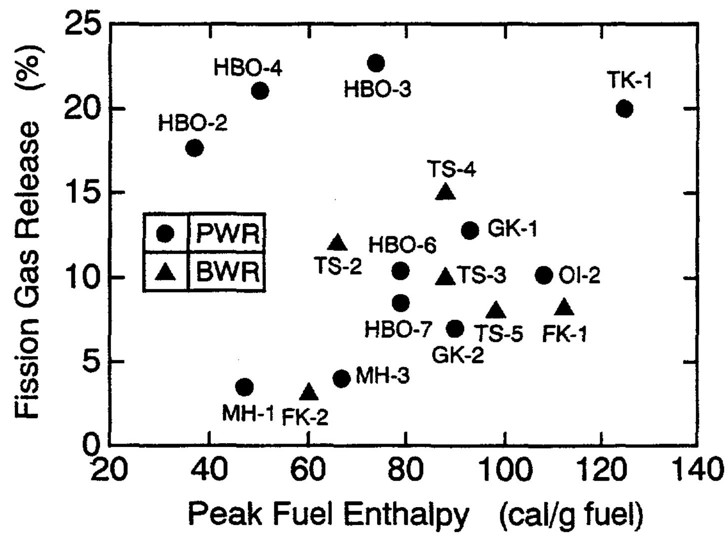

Fig. 19 Fission gas release in PWR and BWR fuel experiments

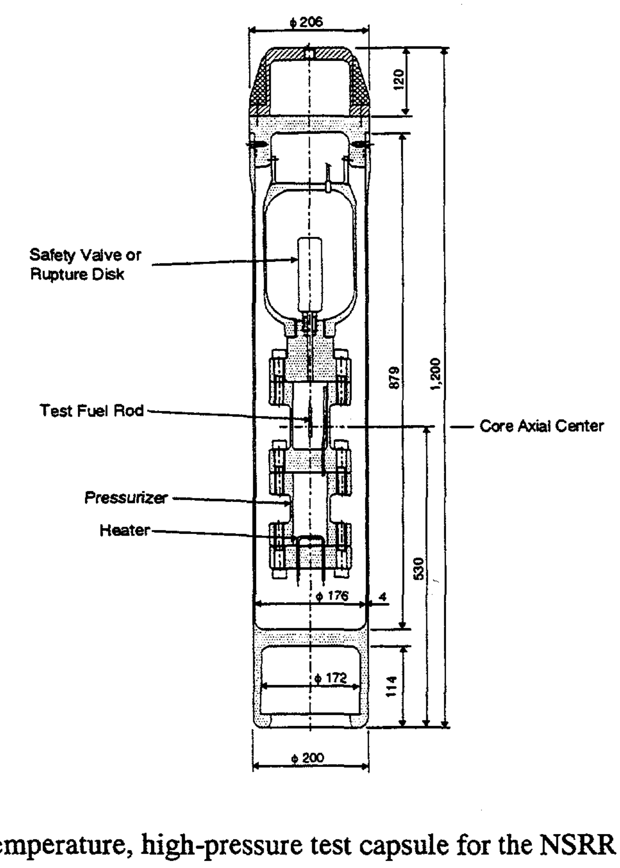

Fig. 20 High-temperature, high-pressure test capsule for the NSRR experiment

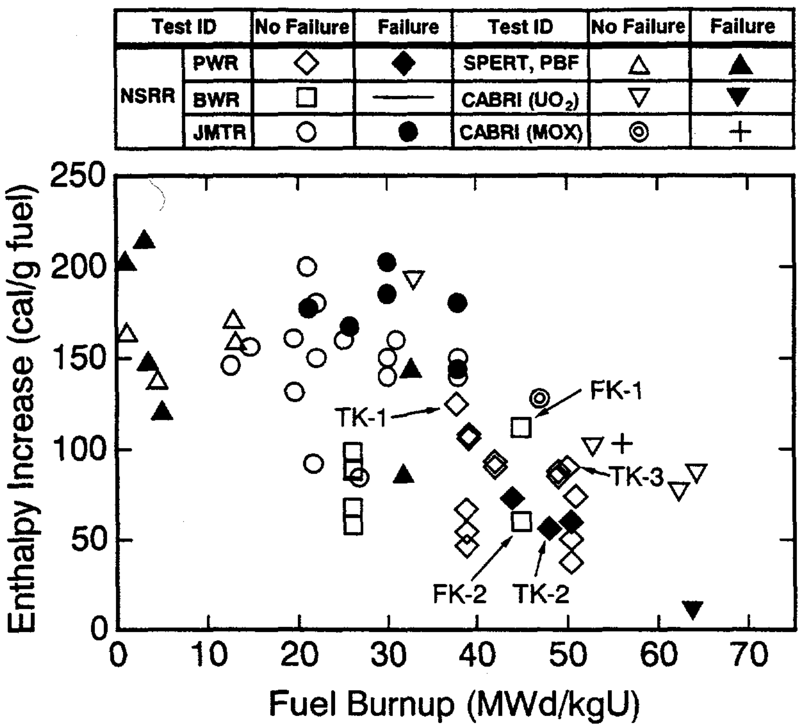

Fig. 21 Existing data from in-pile RIA experiments

THE STATUS OF THE CABRI - REP-Na TEST PROGRAMME : PRESENT UNDERSTANDING AND STILL PENDING QUESTIONS

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ABSTRACT

All the experiments of the original CABRI-REP Na test matrix (6 UO_2 and 3 MOX tests) are performed presently. The last three tests have been conducted during the first half of this year. All three test devices must still undergo destructive examinations, therefore final conclusions cannot be formulated yet. Nevertheless it is possible, at this stage, to present a provisional picture of the fuel behaviour understanding based on the full spectrum of our investigations. In particular, by using the results of SCANAIR code calculations which are compared to the data of the PROMETRA tests, it is possible to quantify, in a first approach, the failure risk of each of the CABRI tests and to evaluate the influence of the various test parameters.

Within the UO_2 tests, the striking difference between REP-Na1 (failure) and the good performance observed in REP-Na8 reveals the importance of the energy injection rate (pulse width). A burn-up enhanced MOX-effect is to be postulated in order to explain the failure of REP-Na7 with regard to comparable UO2 experiments and to the two other MOX tests.
Fission gas dynamic loading is most probably the underlying key phenomenon explaining both observations, the ramp-rate-effect and the MOX-effect.

The CABRI REPNa programme has provided important new knowledge on the behaviour of high burn-up fuel during the early phase of the RIA transient.
However neither NSRR nor CABRI REP Na programmes allow presently to establish the correct boundary conditions of the coolant temperature velocity and pressure and to reproduce the reactor-representative pressure gradient between fuel and coolant.

I - INTRODUCTION

The steadily increasing concentration of fission products and other irradiation induced high burnup phenomena like the RIM-effect, together with longterm corrosion phenomena (oxide layer, hydrogen pickup, spalling) lead to substantial changes of the fuel of the Pressurized Water Reactors (PWR). One aspect is the characteristic difference with regard to the as-fabricated fuel, the other point being that it cannot anymore be postulated that all the fuel pins at a given burnup level are identical e.g. characterized just by the value of burnup.

The need to demonstrate the validity at high burnup of safety criteria, which were established on basis of experimental investigations with fresh and moderate burnup fuel and known to be conservative, was at the origine in 1989 to start a broad research and development program in the field of reactivity initiated accidents (RIA).

Separate effect tests, global experiments and finally elaboration and validation of computer models are the constituants of this large program which is performed in close cooperation between the french nuclear industry (EDF,FRAMATOME) and IPSN in the facilities of IPSN and of the Commissariat à l'Energie Atomique (CEA).

Early results of the japanese NSRR experiments gave important hints for high burnup effects, in particular the risk of early fuel-pin failure by pellet to clad mechanical interaction (PCMI) which is inoperative at low burnup. This finding was the trigger to launch the CABRI-REP.Na programme intending to provide further global in-pile test results and to produce complementary information with regard to the NSRR tests. The detailed objectives and the results of the first six tests have been presented to the water-reactor-safety community at the occasion of the WRSM since 1993 [1-4].

All the experiments of the original CABRI-REP Na test matrix (6 UO_2 and 3 MOX tests) are perfomed presently. The last three tests have been conducted during the first half of this year. All three test devices must still undergo destructive examinations, therefore final conclusions cannot be formulated yet. Nevertheless it is possible, at this stage, to present a provisional picture of the understanding which is based on the full spectrum of our investigations. In particular, by using the results of SCANAIR code calculations which are compared to the data of the PROMETRA tests, it is possible to quantify, in a first approach, the failure risk of each of the CABRI tests and to evaluate the influence of the various test parameters.

Within the UO_2 tests, the striking difference between REP-Na1 (failure) and the good performance observed in REP-Na8 reveals the importance of the energy injection rate (pulse width). A burn-up enhanced MOX-effect is to be postulated in order to explain the failure of REP-Na7 with regard to comparable UO2 experiments and to the two other MOX tests.
Fission gas dynamic loading is most probably the underlying key phenomenon explaining both observations, the ramp-rate-effect and the MOX-effect.
The CABRI REPNa programme has provided important new knowledge on the behaviour of high burn-up fuel during the early phase of the RIA transient.
However neither NSRR nor CABRI REP Na programmes allow presently to establish the correct boundary conditions of the coolant temperature velocity and pressure and to reproduce the reactor-representative pressure gradient between fuel and coolant.
Despite considerable progress in the understanding of high burnup phenomena during RIA transients it must therefore be stated that more representative experiments are needed in order to reach the program goal and to fully validate the computer models for reactor application.

II - OVERVIEW OF THE NEW CABRI REP NA RESULTS

The CABRI REP Na experimental programme consists of nine tests (six tests with UO_2 fuel and three tests with MOX fuel rods) the parameters being :

- the burn-up level (28 Gwd/t to 64 Gwd/t)
- the degree of clad corrosion (4 to 130 µm ZrO_2) with possible initial spallation and associated hydride concentration and redistribution
- the transient energy deposition (95 to 228 cal/g)
- the pulse half width (10 to 80 ms) leading to different energy injection rates.

The first five tests with UO_2 fuel rods have been already presented in previous papers ([2] to [6]) and have shown the possible occurrence of early rod failure by hydride assisted PCMI in case of high clad corrosion level and high burn-up fuel, while the residual strains of the unfailed rods confirmed the PCMI loading. Fuel fragmentation, large fission gas release and transient oxide spallation were evidenced.

Since last year, three experiments have been performed in the CABRI REP Na loop : the two MOX fuel tests REP Na7 and REP Na9 in complement to REP Na 6, the first MOX fuel test, and the UO_2 fuel test, REP Na8.
The table 1 gathers the main characteristics and results of all the tests.

TABLE 1 : The CABRI REPNa Tests

UO_2 Fuel

Test	Rod	Pulse (ms)	Energy end of peak (cal/g)	Corrosion (μ)	RIM (μ)	Results and observations
Na-1 (11/93)	GRA 5 4.5 % U 64 GWd/t	9.5	110 (at 0.4 s)	80 initial spalling	200	**- Failure,** brittle type for **H_f = 30 cal/g** **- Hydride accumulation** **- Fuel dispersion 6 g,** including fuel fragments outside RIM (> 40 μ) - Pressure peaks in Na of 9-10 bars
Na-2 (6/94)	BR3 6.85 % U 33 GWd/t	9.5	211 (at 0.4 s)	4		**No failure** **Hmax = 210 cal/g** Max. strain : 3.5 % average FGR : 5.5 %
Na-3 (10/94)	EDF 4.5 % 53 GWd/t	9.5	120 (at 0.4 s)	40	100	**No failure** **Hmax = 125 cal/g** Max. strain : 2 % FGR : 13.7 %
Na-4 (7/95)	GRA 5 4.5 % U 62 GWd/t	# 75	95 (at 1.2 s)	80 no initial spalling	200	**No failure** **Hmax = 99 cal/g** **Cladding spalling under transient** Max. strain : 0.4 % FGR : 8.3 %
Na-5 (5/95)	GRA 5 4.5 % U 64 GWd/t	9.5	105 (at 0.4 s)	20	200	**No failure** **Hmax = 115 cal/g** Max. strain : 1 % FGR : 15.1 %
Na-8 (07/97)	EDF 4.5 % 60 GWd/t	≈ 80	106 (at 1.2 s)	130	200	**No failure** **Hmax = 109 cal/g** Spalled rod

MOX Fuel

Na-6 (03/96)	MOX 3 cycles 47 GWd/t	35	126 à 0.66 s 165 at 1.2 s	40		**No failure** **Hmax = 145 cal/g** Max. Strain : 3.2 % FGR : 21.6 %
Na-7 (1/97)	MOX 4 cycles 55 GWd/t	40	125 à 0.48 s 175 à 1.2 s	50		**Failure,** **H_f= 120 cal/g** Strong flow ejection, pressure peaks of 200-110b, fuel motion in the lower half zone
Na-9 (04/97)	MOX 2 cycles 28 GWd/t	34	211 at 0.66 s 228 at 1.2 s	< 20		**No failure** **Hmax = 200 cal/g**

All the three tests have been realised with reconditioned rods from PWR fuel (fissile length 56 cm, rod filled with Helium under 3b) cooled by sodium flow (inlet temperature of 280 °C, velocity of 4 m/s, 2b pressure) and starting from initial zero power.

All the MOX fuel rods are issued from the MIMAS fabrication process, producing $UPuO_2$ agglomerates of 30 % Pu initial enrichment, surrounded by a depleted UO_2 matrix. The mean size of the agglomerates is initially 20 µm, a fraction of them being possibly larger (less than 2 % up to 400 µ) which induces a fissile material heterogeneity mainly for fresh or low irradiated MOX fuel.

The REP Na 7 test has been performed using a 4 cycles MOX fuel rod (55 Gwd/t) reconditioned from the 5th span of a PWR rod irradiated in Gravelines 4 power plant and with a clad corrosion thickness of 50 µm.

The neutron-radiography did not exhibit any hydride accumulation (so called « blister ») nor spalling of the oxide layer.

The power transient of 40 ms half width led to rod failure (at 453 ms) for an injected energy of 109 cal/g at peak power node (PPN).
According to calculations with the SCANAIR code [7] the rod failed at the time when a mean fuel enthalpy of 120 cal/g at PPN was reached (fig. 1).
The failure was immediately followed by a strong sodium flow ejection and high pressure peaks in the channel (200 b at inlet, 110 b at outlet) and by the voiding of the coolant channel (fig. 2).

From the microphones and flowmeter signal analysis, the failure has been located around PPN (26 cm from bottom of fissile length). However, the hodoscope did not give any evidence of fuel motion at that time due to its low sensitivity with low enriched PWR fuel.

A second event, in the lower part of the test rod, occurred 18 ms later (seen by hodoscope, flowmeter, pressure transducers), clearly indicating fuel motion in the lower part of the channel : at this time, which could be considered as the latest one for the onset of fuel ejection, the maximum fuel enthalpy is evaluated to be 130 cal/g.

The important amount of fuel motion up to the end of the test is confirmed by the low residual sodium flow (5 % of its initial value) indicating an almost complete channel blockage. This point is corroborated by the non-destructive examinations showing loss of fuel in the lower part of the fissile column and relocation at the levels of the filters.

At the present time no additional information from post-test examinations is available due to the delay of hot cells work.

The REP Na 9 test has been performed using a 2-cycles MOX fuel rod irradiated in St Laurent B1 power plant. The test rod was reconditioned from the 5th span of an industrial rod with 28 Gwd/t burn-up and a low degree of clad corrosion ($\approx$ 10 µm ZrO_2)
The power transient of 34 ms half width did not lead to rod failure although the high energy injection (228 cal/g at 1.2 s) resulted in a maximum mean fuel enthalpy of 200 cal/g.

The maximum fuel and clad elongations are 8 mm and the residual clad elongation amounts to 5 mm. The transient evolution of the sodium flow rates showed rapid variations (« TOP effect ») due to the transient radial deformation of the rod and to the thermal expansion of the sodium and outer wall.

As for REP Na 7, no information from post-test examinations is available.

In complement to the first high burn-up UO_2 tests, an additional experiment **REP Na 8 has been** performed.

Its objective was to investigate on a high burn-up fuel rod (60 Gwd/t) with high clad corrosion level (130 µ ZrO_2), the effect of the presence of some spallation of the oxide layer under a slow power pulse typical of the reactor case (half width > 40 ms). It is to be compared to REP Na 4 (unfailed rod, no oxide spalling, 80 µ of oxide thickness) with regard to the corrosion state and with REP Na 1 with regard to the energy injection rate.

The test rod was reconditioned from the 5th span of a 5-cycles UO_2 fuel rod irradiated in Gravelines 5 power plant. Some spalling of the oxide layer was evidenced, limited to a narrow azimuthal zone (generating line). The presence of hydride spots has been visualized by the neutron-radiography.

The "slow" power transient has injected 96 cal/g at 0.6 s and 106 cal/g at 1.2 s (half width evaluated to 80 ms) which led to maximum mean fuel enthalpy of 109 cal/g.

No clear evidence of rod failure has been obtained although an acoustic event and a moderate but still significant flow rate variation have been registered during the transient as resulting from a possible gas escape through a micro-crack (fig. 3). There was no DND signal detected (detection of delayed neutron emitters) which normally indicates fuel-rod failure. The checking of the rod tightness which is foreseen in the future post test examinations will clarify this point.
However, no fuel ejection nor fuel motion has been detected.

III. UNDERSTANDING OF THE REP Na TESTS WITH REGARD TO ROD FAILURE RISK

Apart from the REP Na 1 and REP Na7 tests which led to rod failure, the REP Na tests already examined, led to clad straining resulting from the rapid transient heat-up of the fuel with contribution of the thermal expansion and fission gas induced swelling to the clad mechanical loading.

Such a clad straining is well correlated to the energy deposition as confirmed by the axial profile of the mean clad plastic deformation following the axial power profile.
The interpretation of the first unfailed UO_2 rods with the SCANAIR code has shown satisfying agreement between measurements and calculated results concerning the maximum mean clad deformation and the maximum transient clad elongation [6].
However no evaluation of the failure risk could be done at that time due to the lack of any realistic data base for clad mechanical properties.

On the other hand, the early rod failure in the REP Na 1 test has underlined the possibility of stress generation at a time when the cladding is still cold and has evidenced the role of spalling of the zirconia layer on the clad embrittlement as widely confirmed by the post-test examinations of the failed rod.

Indeed, a high burn-up rod with a thick outer $Zr\,0_2$ layer of 80 - 100 µm is characterised by a mean hydrogen content of 700 - 800 ppm which results in hydride platelets oriented along the circumference ; if during the base irradiation some oxide spalling occurs, part of the hydrogen migrates towards the outer colder spots of the clad, leading to « blister » formation with locally high embrittlement and sites for crack initiation.

All these considerations were taken into account in the definition of the out of pile PROMETRA test programme whose aim was to provide a data base for the mechanical properties of irradiated cladding under fast transients (strain rates ranging from 0.01 s^{-1} to 5 s^{-1}) to be used in the SCANAIR code.

This programme described in detail in [8], has been realised by performing tensile tests in transverse and rolling directions and with two burst tests, the different parameters being : the strain rate, the temperature (20 - 1100 °C) the corrosion thickness (0µ - 20 µ - 50 µ - 85 µ) and the state of corrosion (with or without spalling).

Although the obtained results show some scattering (which remains to be analysed more in details), the main outcome of this programme can be summarised as follows (fig. 4) :

- the yield stress (YS) and ultimate tensile strength (UTS) are increased with increased strain rate and irradiation and with decreasing T,

- the increased corrosion leads to reduced uniform elongation (UE) and resistance (YS and UTS) compared to less corroded samples, especially at high strain rate,

- in terms of uniform elongation, non negligible ductility has been found even for highly corroded cases (UE ≥ 2 % in axial tests, UE ≥1 % in transverse tests) without evidence of fully brittle result at any temperature and $\dot{\varepsilon}$ even with spalled samples,

- the total elongation of spalled samples may be strongly reduced down to UE values,

- the brittleness is mainly related to the presence of hydride accumulation whose distribution is not fully characterised,

- no striking anisotropy has been found between the hoop and axial tests.

-the two burst tests performed at 350 °C and low strain rate (0.015 s^{-1}) on a severely corroded but not spalled clad (≈ 80µ, ZrO_2) have led to lower UE (0.2 %) and UTS than those issued from uniaxial tests ; such results may be considered as more representative of clad loading by bi-axial stresses resulting from fission gas pressure.

III.1. Analysis of the REP Na tests based on SCANAIR code and PROMETRA results

All the REP Na tests have been analysed with the SCANAIR code giving the evolution versus time of the clad temperature, deformation and strain rate.
In order to evaluate the situation of the tests with regard to failure risk, a criterion has been elaborated based on the comparison of the maximum plastic clad hoop strain (at mid clad thickness) to the possible failure domain between uniform elongation (UE) and total elongation (TE) :

$$FR = \frac{\varepsilon_{\max} - UE\,(T_{\varepsilon_{\max}})}{TE\,(T_{\varepsilon_{\max}}) - UE\,(T_{\varepsilon_{\max}})}$$

The variation of UE and TE with temperature is deduced from the PROMETRA results (linear or constant laws) at given strain rate and corrosion level.
Such formulation allows to consider the different clad loading modes :

- with pure PCMI (imposed deformation) the failure limit is the total elongation
- in case of gas pressure loading, the failure limit corresponds to a stress loading to be compared to the UTS associated to the uniform elongation.

So, the proposed criterion may lead to the following cases :

FR < 0	- absence of failure risk
0 < FR < 1 :	- the failure risk is low in case of pure PCMI with clad straining lower than TE - the failure risk exists in case of pressure loading
FR > 1 :	- the margin to failure is overpassed

The figure 5 illustrates the evaluation of the FR coefficient for all the REP Na tests (except REP Na 1 which is discussed later).

We can notice that the non failure of REP Na 2, REP Na 3, REP Na 5, REP Na 6 and REP Na 9 rods confirms a clad loading by pure PCMI (UE < ε max << C TE)

On the other hand the non failure of REP Na 4 is consistent with the evaluation of absence of risk due to low deformation induced by the slow power transient (as shown on figure 6a by the evolution versus time of the mid clad plastic hoop strain compared to PROMETRA results at the corresponding mid clad temperature).

Concerning REP Na 8 test, a clear determination of the FR value is difficult because the failure domain between TE and UE is limited as the result of the spalling of the oxide (PROMETRA results) and because the mid clad straining is calculated to be very close to the lower limit (UE) as illustrated on figure 6b. Moreover, a SCANAIR calculation has shown that an increase of energy injection of only 6 % would have led to reach the failure domain. Such a result indicates that any small change in the evolution of the clad deformation would have led to enter into the failure zone with a high probability of failure.

The determination of the FR value for REP Na 1 is in fact impossible because it was found from the analysis that the early rod failure occurred within the elastic domain.
However, if we consider the PROMETRA results of the burst tests realised on highly corroded cladding (80 μ oxide, without spalling) which indicate a very low UE (0.2 %) the REP Na 1 clad failure can be interpreted as the result of fission gas pressure loading.
This confirms the assumption of contribution of the rim zone with high gas retention and gas over pressure due to rapid over heating [6, 8].

Concerning REP Na 7 test, we can notice on figure 5 that the FR value is the highest one, consistently with the occurrence of rod failure.
The figure 7 shows the evolutions of the mid clad temperature, plastic hoop strain and strain rate versus time together with the PROMETRA values for the mid-corrosion level (50 µm of $Zr\ O_2$ layer). At the time of rod failure, the calculated clad straining by SCANAIR is 1.2 % (mean clad temperature 534 °C, $\varepsilon = 0.2\ s^{-1}$) overpassing the UE value.
It is to be recalled that the SCANAIR calculation for MOX fuel assumes an homogeneous fuel behaviour.
Such hypothesis seems to be justified by the recalculation of the REP Na 6 test (3 cycles , 47 Gwd/t) giving a maximum clad straining of 2.9 % in agreement with the average measured value of 2.65 %.
On the other hand, in REP Na 7, the axial location of the first failure at peak power level tends to eliminate any effect of fuel heterogeneity due to $UPu0_2$ agglomerates. Moreover, comparing REP Na 7 to REP Na 6 with similar power pulse leading to a maximum fuel enthalpy of 145 cal/g without failure, would suggest the influence of burn-up on the clad loading.
Indeed, in the MOX fuel, the high concentration of fission gases in the $UPu0_2$ agglomerates over the whole section can be compared to the local rim zone of high burn-up UO_2 fuel and may be responsible

for gas overpressure (the agglomerates represent roughly 20 % of the fuel mass compared to 5 % for the rim zone of a 5-cycles UO_2 rod).
Clad loading with contribution of fission gas pressure can explain the occurrence of rod failure with a clad deformation close to UE and so suggests a MOX effect at high burn-up. The high level of confinement and concentration of gases could also explain the high flow ejection at failure time.
However, the precise description of the MOX fuel behaviour by the SCANAIR code is not validated. Additional information from future post-test examinations is needed for better understanding.

III.3. Influence of the power ramp rate effect

The study of the influence of the power ramp rate effect on the rod behaviour was one of the objectives of the REP Na8 test (60 Gwd/t, 130 μ oxide thickness with some spallation, pulse half width of 80 ms) in comparison to REP Na 1 test (63 Gwd/t, 80 μ oxide thickness, spallation, pulse half width of 10 ms).

The result of REP Na 8 test without clear evidence of rod failure suggests a benefic effect of a slower pulse already found by the non failure of REP Na 4 test (rod without oxide spalling).
Indeed, with such a slow pulse the enthalpy increase rate (maximum mean fuel enthalpy) is almost ten times lower than in REP Na 1 test (1.28 cal/g/ms compared to 10 cal/g/ms in REP Na 1).
Parametric calculations with the SCANAIR code considering different pulse half widths (80, 60, 40, 20, 10 ms) for the same energy injection have shown that the fastest the pulse is, the highest mean fuel enthalpy, clad strain rate and strain energy density are obtained in relation with a more adiabatic behaviour of the fuel during the pulse.
It is clearly shown that the REP Na 8 rod submitted to a power pulse similar to REP Na 1 would have reached clad deformation inside the failure domain as defined in III.1, with a high probability of failure (ε_{max} > UE).
On the other hand, under a fast power pulse, the gas dynamic behaviour is much more effective due to rapid over pressure of the inter-granular and porosity bubbles leading to fuel fragmentation and contribution to clad loading as deduced from REP Na 1 analysis : together with the low ductility of a spalled cladding, rod failure is more likely.

In order to better understand and to verify these points, the possibility of performing an additional test with UO_2 high burn-up fuel is currently examined in the REP Na test matrix **(REP Na 10)**.
For better comparison to the available results, a 5-cycles rode similar to REP Na 1 rod would be chosen (80 μ oxide layer with spallation) with a power pulse of 30 ms half width which can be considered as a penalising ramp in a reactor case. Such test should complement the knowledge by comparison to REP Na 1 (fast pulse) and REP Na 8 (slow pulse).

IV. PENDING QUESTIONS AND FUTURE NEEDS

The recent experiments in CABRI and NSRR have given evidence that the RIA safety criteria are not longer valid at high burn-up. These criteria were based on the overpassing of the critical heat-flux (DNB) as postulated failure condition and on the onset of fuel melting as initial condition for fuel dispersal.

At high burn-up the picture is fundamentally changed :

- fission gas driven fuel fragmentation provides micron or submicronsized fuel grains which are promptly ejected and dispersed when pin rupture occurs.
- DNB occurrence is strongly promoted as a result of the fuel/clad-gap closure at high burn-up and due to the high contact pressure, both phenomena increase strongly the heat fluxes under RIA conditions.

Pin rupture might result from PCMI or from the combined damage of early PCMI and subsequent DNB effects (decreased clad mechanical resistance, still low ductility).

The results obtained from the REP-Na1 and REP-Na7 tests are to be explained by dynamic, undelayed gas effects from the RIM region and from the MOX-clusters respectively and direct loading to the cladding. In CABRI this loading mode may be mitigated due to the low pressure conditions. The NSRR tests GK1 and GK2 gave hints that high internal pressure enhances the gas release and increase the clad straining when the internal pin-pressure overreaches the coolant pressure. This would be the case in the reactor conditions with an initial internal pin pressure of about 100b at the end of a 5-cycles irradiation to be compared to 3b in CABRI REP-Na.

High fission gas release is also observed in CABRI and in NSRR when clad straining provides space for efficient fuel fragmentation and porosity opening.
The kinetics of the different release modes are not known presently and the sequence of these phenomena and the influence of the surrounding pressure cannot be simulated with a validated version of the SCANAIR code.
Separate effect tests for better understanding of the transient gas behaviour and fuel fragmentation are programmed in the french SILENE reactor using small samples of fuel which are submitted to fast power pulses. Only partial answers however will be obtained from these experiments.

On the other hand, SCANAIR simulations of RIA transients under reactor conditions with a 5-cycles PWR rod (similar to REP Na 4) have shown a high sensitivity to the clad-fluid heat transfer concerning onset of boiling crisis (DNB). For instance using a critical heat flux of 1.8 MW/m^2 or the standard SCANAIR correlation (minimum temperature of film boiling from Groeneveld-Stewart law) leads to onset of DNB for mean fuel enthalpy of 60 or 95 cal/g.
The out of pile PATRICIA programme devoted to the determination of heat exchange laws and presently underway, should reduce the uncertainty in this field but nevertheless will not address the whole fuel rod behaviour after boiling crisis (fuel and clad behaviour).
Another pending question is the effect of the transient oxide spallation evidenced in several REP Na tests (REP Na 3, REP Na 4, REP Na 6). SCANAIR simulation of this phenomenon has shown that due to increased heat transfer to the fluid, clad temperature increase is accelerated leading to early DNB (at H = 80 cal/g in case of a pulse of 20 ms halfwidth).

The previously presented procedure for failure risk evaluation represents significant progress but cannot be considered as a provisional pin-rupture criterion. The use of the UE value as failure criterion would be too conservative for still ductile cladding. It would be insufficiently precise for brittle cladding and eventually incorrect as seems to indicate the result of REP-Na1.
So it must be concluded that both the transient clad-loading phenomena and the mechanical cladding response are not sufficiently understood presently.

Future experiments under representative PWR conditions are needed in order to cover the full range of the sequence of phenomena :

- combined effect of early PCMI and subsequent DNB at low mean fuel enthalpy,
- fission gas release and dynamic loading action at high system pressure,
- fuel fragmentation and pin failure,
- post failure fuel motion and fuel coolant interaction.

The implementation of a pressurised water-loop into the CABRI reactor with its capabilities for variable energy injection rates and with the present and in future improved energy, deposition capabilities would provide the experimental facility allowing to address and resolve the still open questions and to provide, for the future fuel optimisation, all possibilities for representative experiments, with special emphasis on fast transient heating but allowing also to perform studies in a limited field of the LOCA scenario.

V. CONCLUSION :

The original CABRI REPNa test programme has been realised, with the last three tests being performed this year and has given highlights on the fuel rod behaviour during the first phase of a RIA transient with negligible clad heat-up.

Although all the post-test examinations are not available due to delay of hot cells work, a general understanding could be derived.

The analysis of the tests based on SCANAIR code and PROMETRA results for clad mechanical properties, led to establish a failure risk scale.
The benefic effect of a slow power pulse (80 ms halfwidth compared to 10 ms) on a high burn-up fuel rod is evidenced through REP Na 4 and REP Na 8 results in comparison to REP Na 1 in which the role of the gas dynamic behaviour of the rim zone on the clad loading can be confirmed.

An additional test, REP Na 10 with high burn-up fuel, spalled clad and power pulse of 30 ms half width would complement this set of experiments in the near future.

On the other hand, the failure of the MOX fuel rod REP Na 7 suggests a high burn-up effect with contribution of fission gas pressure from the $UPuO_2$ agglomerates.

However at the present time, pending questions are still open such as the rod behaviour after PCMI phase with significant clad heat-up (up to and beyond DNB) the kinetics of gas release mainly with a high internal pressure, the post failure phenomena.

More representative experiments in a pressurised water loop inside the CABRI reactor would allow to study and quantify the whole sequence of a RIA providing physical knowledge for the validation of the computer codes and reliable reactor application.

ACKNOWLEDGMENTS :

The authors express their acknowledgements to M. BALOURDET, B. CAZALIS, H. RIGAT who contributed to the present paper and to the CABRI team who realised successfully the CABRI REP Na experiments.

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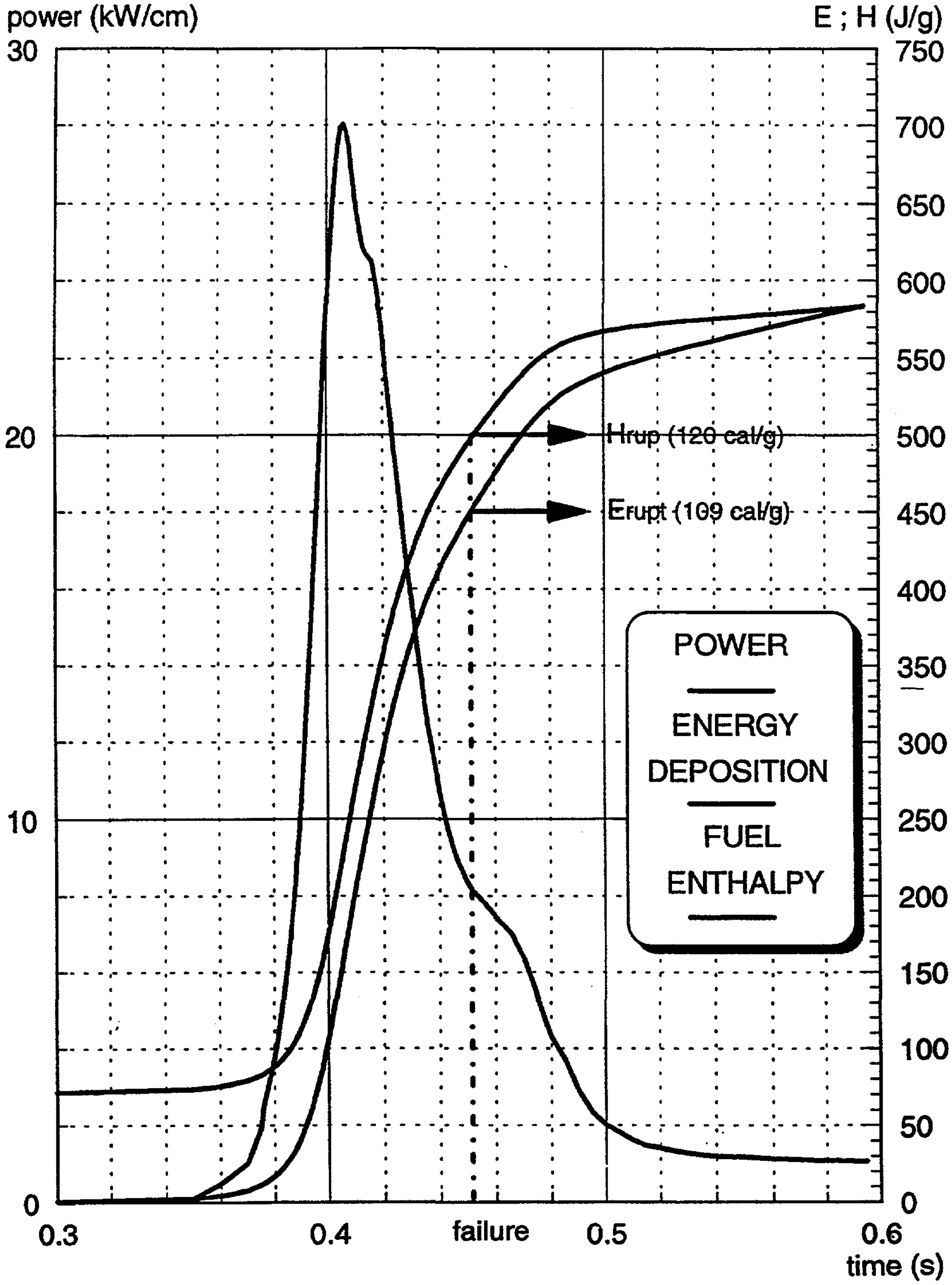
FIGURE 1 : REP Na 7 : POWER, ENERGY, FUEL ENTHALPY AT PEAK POWER LEVEL VERSUS TIME (SCANAIR CALCULATION)



<u>FIGURE 2 :</u> FLOW RATE EJECTIONS IN THE TEST CHANNEL CONSECUTIVE TO ROD FAILURE IN REP Na 7

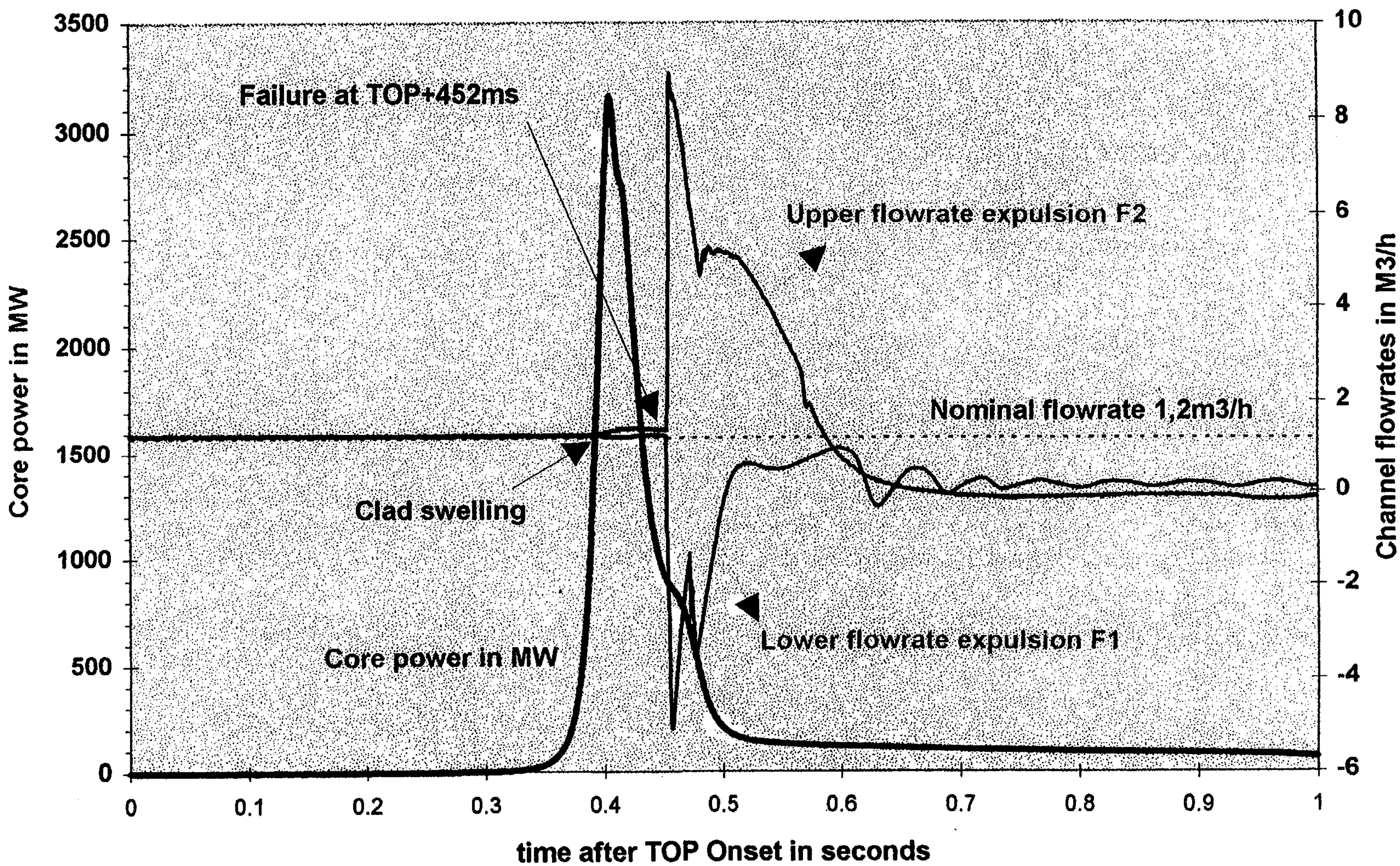
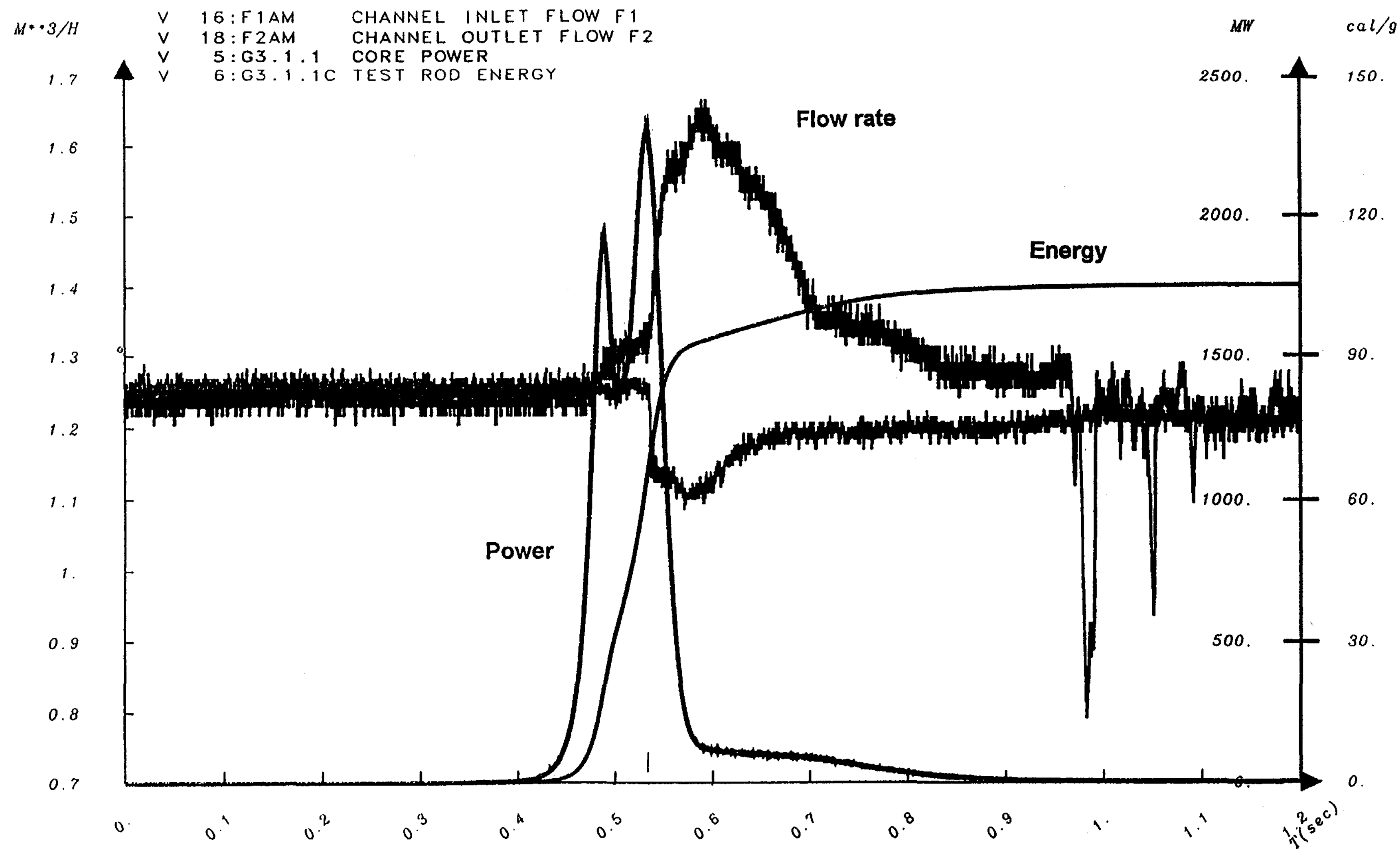


FIGURE 3 : POWER, ENERGY AND FLOW RATE EVOLUTION VERSUS TIME IN REP Na 8



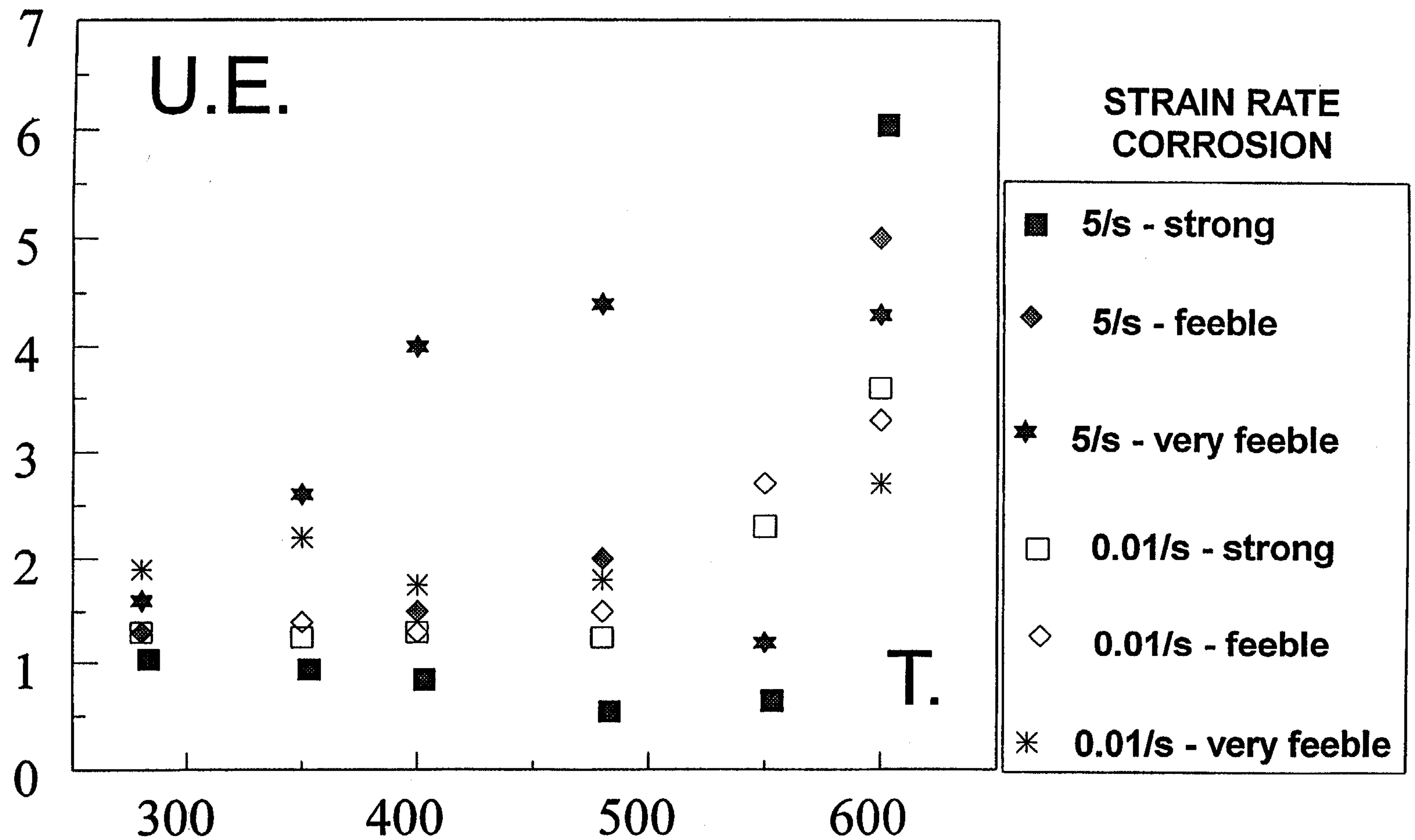


FIGURE 4 : UNIFORM ELONGATION IN THE TRANSVERSE DIRECTION VERSUS TEMPERATURE : PROMETRA RESULTS

**<u>FIGURE 5</u> : RISK OF FAILURE OF CABRI REP Na TESTS
EVALUATION FROM SCANAIR AND PROMETRA RESULTS**

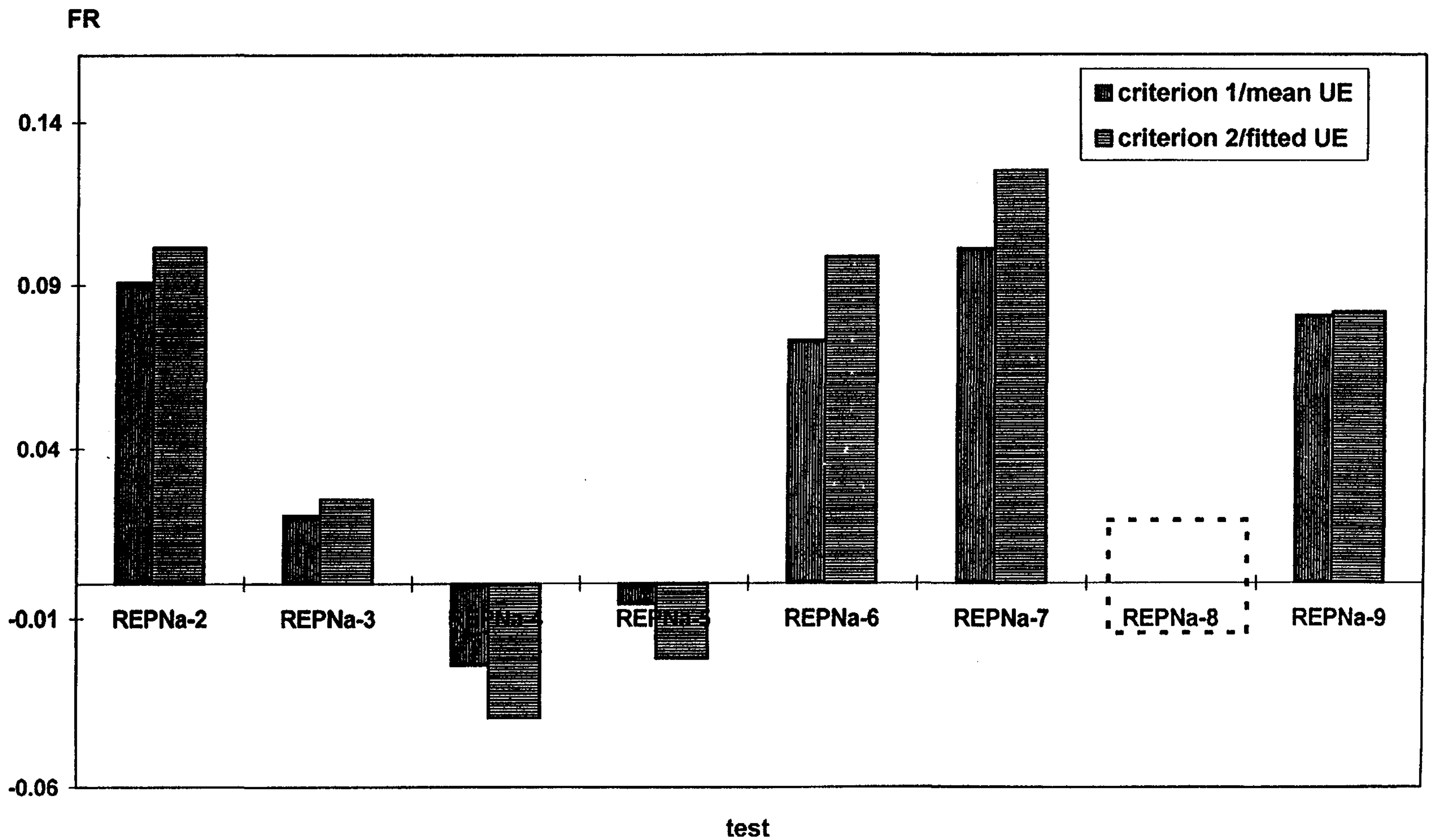


FIGURE 6 a

REPNa-4
evolution of mid-clad plastic hoop strain versus time compared to PROMETRA results

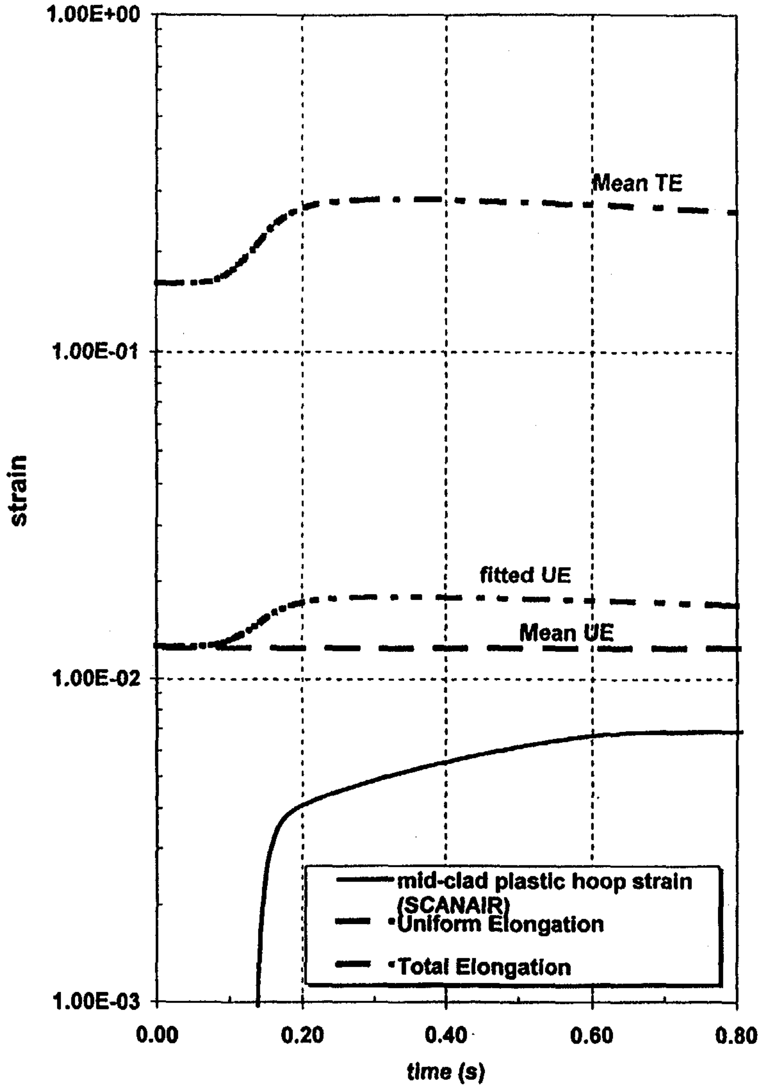


FIGURE 6b

REPNa-8
evolution of mid-clad plastic hoop strain versus time compared to PROMETRA results

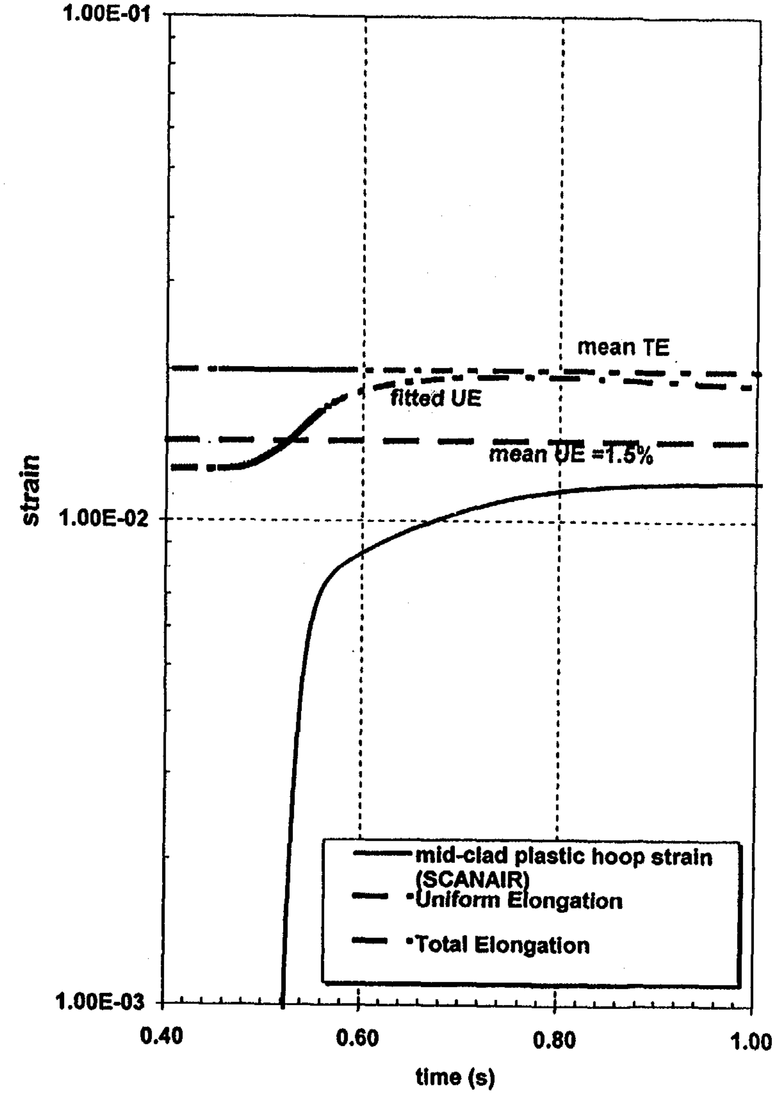
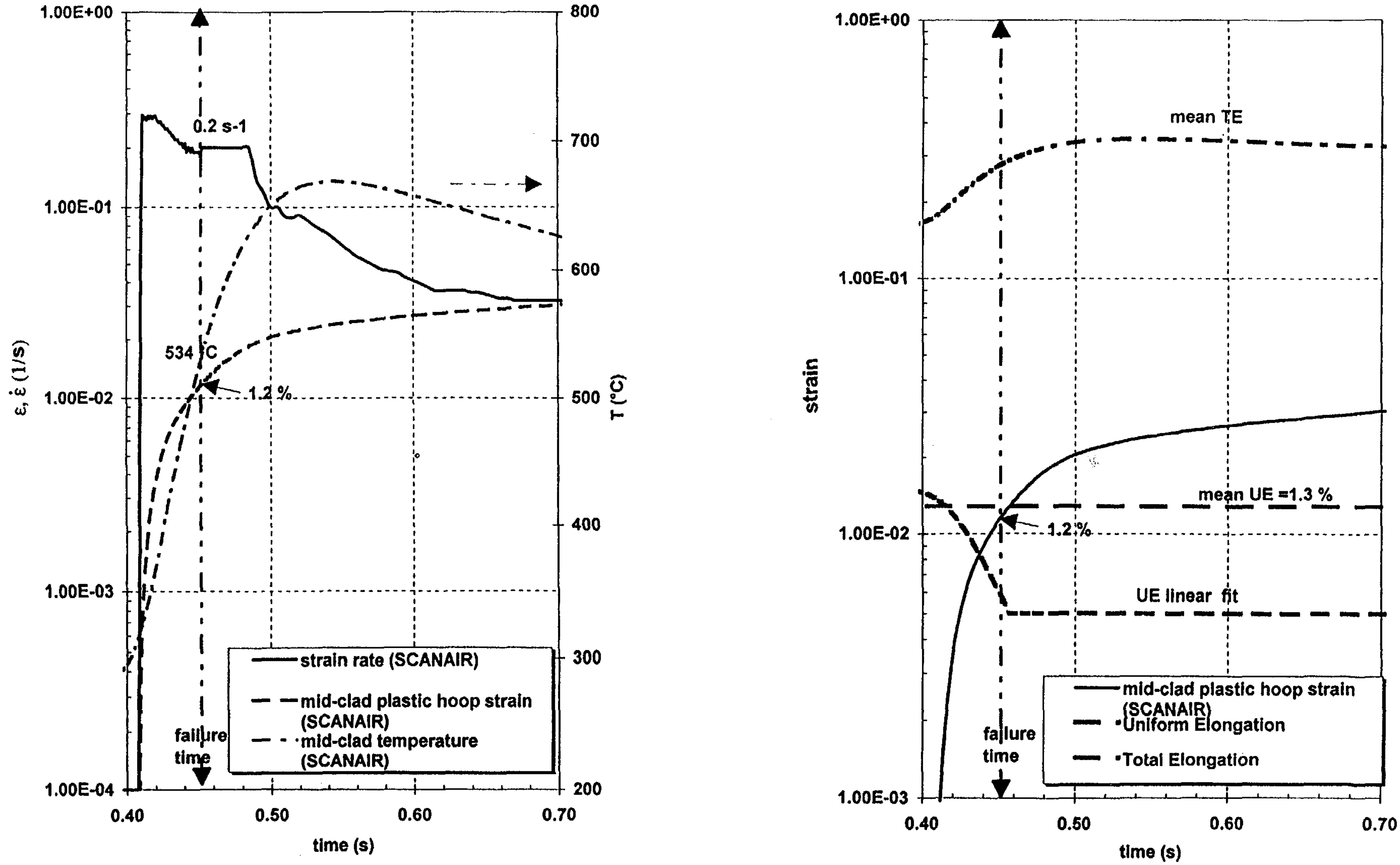


FIGURE 7 :

REPNa-7
evolution of mid-clad plastic hoop strain, strain rate and mid-clad temperature versus time

REPNa-7
evolution of mid-clad plastic hoop strain versus time compared to PROMETRA results



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10. SUPPLEMENTARY NOTES

C. Bonsby, NRC Project Manager. Proceedings prepared by Brookhaven National Laboratory

11. ABSTRACT *(200 words or less)*

This three-volume report contains papers presented at the Twenty-Fifth Water Reactor Safety Information Meeting held at the Bethesda Marriott Hotel, Bethesda, Maryland, October 20-22, 1997. The papers are printed in the order of their presentation in each session and describe progress and results of programs in nuclear safety research conducted in this country and abroad. Foreign participation in the meeting included papers presented by researchers from France, Japan, Norway, and Russia. The titles of the papers and the names of the authors have been updated and may differ from those that appeared in the final program of the meeting.

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