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Nuclear Safety is a journal that covers significant issues in the field of nuclear safety.

Its primary scope is safety in the design, construction, operation, and decommissioning of nuclear power reactors worldwide and the research and analysis activities that promote this goal, but it also encompasses the safety aspects of the entire nuclear fuel cycle, including fuel fabrication, spent-fuel processing and handling, and nuclear waste disposal, the handling of fissionable materials and radioisotopes, and the environmental effects of all these activities.

Qualified authors are invited to submit articles; manuscripts undergo peer review for accuracy, pertinence, and completeness. Revisions or additions may be proposed on the basis of the results of the review process. Articles should aim at 20 to 30 double-spaced typed pages (including figures, tables, and references). Send inquiries or 3 copies of manuscripts (with the draftsman's original line drawings plus 2 copies and with black-and-white glossy prints of photographs plus 2 copies) to Dr. M. D. Muhlheim, Oak Ridge National Laboratory, P. O. Box 2009, Oak Ridge, TN 37831-8065.

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Editorial Staff

General Safety Considerations

Edited by D. A. Copinger

The Nuclear Community and the Public: Cognitive and Cultural Influences on Thinking About Nuclear Risk

By M. A. Meyer[a,b]

Abstract: *This article examines why the public holds views of nuclear-related risk different from people working in the field of nuclear safety. In particular, the study looks at how feelings enter into thinking about risk. It focuses on (1) the nuclear community, specifically the technical experts who perform accident analyses, and the regulators who use these analyses in making risk assessments or policy decisions; and (2) the general public. This article summarizes these groups' approaches to nuclear risk and explores the effects of cognition and cultural conditioning in creating these differences. The goal is to increase the nuclear community's understanding of the public's approach to risk, as well as its own, in hopes of improving communication.*

This article is a summary of literature gathered from diverse fields describing how people think and feel about nuclear-related risks. Its aim is to help the nuclear community, especially the U.S. Nuclear Regulatory Commission (NRC) and its contractors, understand how they differ from the public in thinking about such risks.

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The intended audience includes the scientists and engineers who work in the area of nuclear safety. They may be technical experts, such as the scientists who perform accident analyses by creating and running complex computer models, or they may be regulators, the decision makers who use these analyses in making risk assessments or policy decisions. The article describes the cognitive and cultural influences that selectively operate to form their views of nuclear risk; however, this study is expected to benefit any member of the nuclear community who communicates with the public about nuclear safety.

Understanding how the public thinks about nuclear risk is important for several reasons. First, nuclear engineers and scientists often have dealings with the public concerning nuclear safety; for instance, the NRC personnel frequently communicate with the public, such as in circulating rules and environmental assessments for review, responding to petitions and allegations, holding open meetings, sponsoring workshops on controversial issues, and answering general questions about nuclear energy and radiation issues. Understanding people's thinking about risk has been identified as necessary to communicating effectively about risk and to creating acceptable public policy.[1,2]

Second, communications with the public on nuclear risk have been problematic; for example, the NRC has encountered difficulties in communicating information to the public; the NRC provides information when individuals ask or when the regulations stipulate (e.g., when environmental assessments are to be disseminated for review). The purpose of the NRC's informational program is to explain the technology, to articulate its risk with precision, and to let people evaluate for themselves the technology's acceptability;[3] however, many times the public does not want to hear the "facts" about nuclear energy, nor do they believe the NRC's assessments of risk. In one well-known instance, H. Denton, the NRC's chief official on site after the Three Mile Island (TMI) accident, could not convince a portion of the public that the radioactive release had been very small—a fraction of the agency's regulatory limit. These individuals would not believe the NRC records of the release. Because they had metallic tastes in their mouths, they were convinced that there had been a massive release.[4]

The work described here began as an attempt to answer some questions concerning people's thinking about risk:

- Why has informing the public about the scientific basis of risk assessments had so little effect on the public's views in the last few decades?
- How do feelings enter into thinking about risk?
- In particular, why are the public's feelings about nuclear risk so resistant to change or scientific counterargument?
- More generally, why do technical experts view risk so differently from the public?

This article offers answers to these questions. It differs from other reviews of the risk literature in three ways: (1) it includes findings from more diverse fields, such as physiology and anthropology; (2) it examines the role of people's emotions in their thinking about risk; and (3) it proposes the cognitive mechanisms by which individuals become aware of risk and deal with it. In general, this article proceeds from describing how different groups evaluate risk to explaining why these differences exist.

Specifically, the article is organized as follows: (1) a description of how the public and the technical experts/regulators approach risk, including recent findings on the public's feelings about nuclear-related risks; (2) an illustration of the physiological, emotional, and cognitive mechanisms involved in the individual's response to risk; (3) a culturally based explanation for the differences in the technical experts' and public's views of nuclear risks; and (4) a summary.

THE PUBLIC EVALUATES NUCLEAR-RELATED RISKS DIFFERENTLY THAN EXPERTS OR REGULATORS DO

To describe the views of risk of different groups, risk must first be defined. Risk is the "potential for realization of unwanted, negative consequences."[5] Negative consequences can range from relative intangibles, such as decreased quality of life (e.g., as the result of mental anguish), to more concrete possibilities, such as the loss of health, life, or property; for instance, making a left turn across traffic could be viewed as risking frustration, loss of time, vehicle damage, injury, or even death.

The Public's Approach to Risk

The public is defined here to mean the diverse groups of citizens, some of whom may belong to special interest groups, such as the Sierra Club. This population is frequently studied by means of random telephone or mail surveys. According to such surveys, the public's approach to risk tends to be qualitative, anecdotal, and personal. Typically, the public thinks about risk in terms of their feelings and of the effects of the risks on themselves and their loved ones.[6]

The following additional characteristics of people's thinking are likely to impact their evaluation of nuclear-related risks.

Individuals mentally lump the risks from nuclear weapons and nuclear waste with those of nuclear power reactors. Evidence that lay people mentally lump all nuclear risks can be found in Slovic's study of risk perception.[7] In this study, people ranked nuclear reactor accidents next to nuclear weapons' fallout in their perception of the riskiness of these hazards.

Additional evidence that people lump nuclear risks comes from psychological studies of images. These studies are based on the concept that people's cognitive images are accompanied by feelings and that such images have important behavioral consequences. In particular, people's images of a city predict their preferences for vacationing or relocating there;[8] for instance, an image of a sunny beach and azure water is likely to encourage tourists, especially in the winter.

In studies of images, people are given a word or phrase such as "reactor accident" and asked for the words they associate with it. In three separate studies of the images associated with reactor accidents, nuclear waste repositories, and nuclear war, people's images of disaster were equivalent. In particular, the people's images of the consequences of reactor accidents were the same as those that they gave for nuclear war. [9]

Slovic[8] offers an explanation of why people's perceptions of reactor accidents are so severe when there have been relatively few fatalities to date. He notes that the early reactor risk assessments were worst-case scenarios causing tens of thousands of deaths and that these received much publicity, such as in the movie "China Syndrome."

Individuals implicitly think about many factors in considering riskiness. Experimental psychologists Slovic, Lichtenstein, and Fischhoff have studied people's perceptions by asking them to rank the risks of well-known hazards (e.g., numerically rate the riskiness of these hazards and the level of regulation they desired for each). They also asked people to rate the hazards with regard to characteristics thought to be important for the way people perceive risk.

Their results[7] showed that people's views of risk tended to differ from their own and experts' estimates of annual fatalities.[a] People rank risks on the basis of such characteristics as how well they understand the problem, how equitably they feel the danger is distributed, how well they can control their exposure, whether they have assumed the risk voluntarily, and how children and future generations are affected by the risks.

Individuals view nuclear hazards as riskier. When several of the characteristics associated with riskiness are grouped, nuclear power is one of the highest scoring hazards. Slovic[7] used psychophysical scaling and multivariate analyses to group the following characteristics: "perceived lack of control, dread, catastrophic potential, fatal consequences, and the inequitable distribution of risks and benefits." The higher a hazard's score in this grouping, the higher the perceived risk, the more that people want its risks reduced, and the more they want strict regulation. Nuclear weapons and nuclear power score highest on these characteristics of riskiness. People's perceptions of the riskiness of nuclear power do not change when they consider its parts—radioactive waste, uranium mining, and nuclear reactor accidents. According to Slovic,[7] these results have been replicated in studies across a wide cross section of the population.

Analysis of nuclear imagery reveals negative images indicative of feelings of revulsion and fear. In their study of the images of nuclear waste repositories, Slovic, Layman, and Flynn[8] found extremely negative images of "dangerous/toxic," "death/sickness," "environmental damage," and "bad and scary." Positive images, such as of "employment" or "money/income," accounted for only a few percent of the images. These images were relatively stable across time (1988 to 1990) and populations (the nation at large and residents of Nevada, Arizona, and California).

The researchers compared these images with those in other studies and concluded that the feelings that underlie these images are of "dread, revulsion, and anger; the raw materials of stigmatization and political opposition." Given that people lump nuclear risks together, this conclusion can be viewed as applying to nuclear reactor accidents also.

Revulsion or fear may not be amenable to logical thinking, even in those aware of this effect. Information on how people act when experiencing revulsion comes from studies of averse reactions.[10] In such studies, college students were asked what they would do in the following situations: drink their favorite juice if a dead cockroach was dipped in and removed; if a dead and sterilized cockroach was dipped in and removed; or if a brand new fly swatter was used to stir it. In these three situations, approximately the same number of students said they would not drink their juice; however, only in the first instance, the dead cockroach, could fear of exposure to germs be the reason for revulsion and refusal. In the last two cases, the students knew that they were being irrational; however, this did not change their feeling of revulsion nor their reluctance to drink a contaminated beverage.

This same kind of reaction occurs in other situations where there is even less chance of contagion; for example, college students are reluctant to wear a sweater that has been sterilized after being worn by a person who has committed moral offenses, such as child molestation. The students know that their feelings and behavior are irrational, but they are unable to do otherwise.

Redelmeier, a physician,[10] states that averse reactions, such as revulsion, can be resistant to change, even if people know that these feelings are irrational and even if they have been given scientific counterarguments. He further proposes that people do not usually volunteer information on averse emotions and may not even be aware that they have

[a]According to Slovic,[7] experts' views of risk correlate highly with technical estimates of annual fatalities.

them. This effect is of special interest given the previous evidence of many people's aversion to nuclear technologies.

People's evaluations of nuclear risk may be influenced by their trust in the managing organizations. Although the evaluation of nuclear risks by people may not be affected by logic, it may be influenced by trust. A study was done of Nevadans' views of the nuclear waste repository project.[11] Multivariate analysis showed that perceptions of economic benefits were not good predictors of opposition to the project but that risk perceptions and trust in repository management were closely linked to the positions of people on the project. The trust of people directly influenced their risk perceptions; this, in turn, had a direct effect on their view of the repository. In other words, the confidence of people in the managers of a technology influenced their perception of a technological risk and the position that they took on it.

The authors of *Public Reaction to Nuclear Waste*[12] confirm the importance of trust to the perception of risk and add that public trust in American institutions has been declining for the last 30 years. They note that no new reactors have been built since the late 1970s and that surveys indicate that the majority of Americans are against building new ones. Along these lines, E. Beckjord, the former Director of the NRC's Office of Nuclear Regulatory Research, has stated that public "confidence in the assurances given by the technical experts that nuclear energy was 'safe' was severely shaken" by the TMI event.[13]

Technical Experts and Regulators' Approach to Risk

Technical experts and regulators, in contrast to the public, take a quantitative and abstract view of risk; for example, risk analysis focuses on identifying the hazards and the means by which people would be exposed; for instance, with respect to reactors, technical experts use computer codes to model the paths by which reactor accidents could occur, the probabilities of their occurrence per year, and their consequences.[14] In fact, probabilistic risk analysts define risk in mathematical terms as the expected number of occurrences (frequency) times the consequences for that occurrence. Consequences are usually figured in terms of human lives lost and health effects as the result of exposure to radiation. Radiation exposure is quantified by determining the amount of radiation received as measured in rems. Regulators, such as those in the NRC, then use the results of these risk assessments in making policy decisions. In instances where safety has been found to be adequate and improvements are being considered, then cost–benefit analyses may also be performed. In cost–benefit analyses, the costs and benefits for reducing a risk, such as by implementing a safety feature, are quantified (e.g., in the number of workdays and money lost or gained).

In summary, the technical experts and regulators describe risk in a language of technical detail, quantities, and generalized costs–benefits.[1,6]

The Result—Adversarial Relations in Risk Regulation

That technical experts and regulators evaluate risk differently than the public is especially evident in the regulatory arena. Risk assessors, managers, and regulators are very aware that the public does not share their views of risk. Morgan summarizes their view:

> Many advocates, such as industry representatives promoting unpopular technology or Environmental Protection Agency staffers defending its regulatory agenda, argue that the public has a bad sense of perspective. Americans, they say, demand that enormous efforts be directed at small but scary sounding risks while virtually ignoring larger, more commonplace ones.[1]

Otway,[15] a noted international risk analyst, has characterized the approach to risk regulation in the United States as adversarial: "Regulations are developed in open confrontation, often with resort to the legal system to settle disagreements." Rowe, another risk analyst, confirms Otway's view and elaborates on this process: issues that are unacceptable to some groups are "blown up through dire predictions of consequences, based primarily on half truths, but flamed by competing commercial news media."[5] The goal is to stir public opinion so as to affect governmental representatives, consumer regulatory agencies, and the courts.

This confrontational approach seems especially true of interactions on nuclear power and radioactive waste disposal. "Decisions that formerly were exercised by scientists, technology managers, and public officials are now subject to extensive public debate, and in many cases decisions are reversed because of the public."[11,13] Two examples of the effect of public opposition are the decline in nuclear power plant construction and the delay in the siting of the nation's first high-level nuclear waste repository.

According to Otway,[15] technical experts have been surprised by the increasingly confrontational atmosphere, especially by the "lay challenges to their informed expert judgment." As scientists they had believed that regulatory decisions would be less controversial if their basis in science could be established.

This situation has been the case in the probabilistic risk and safety assessments (PRAs and PSAs) performed on nuclear power reactors. Analysts in this field have believed, albeit for different reasons, that the scientific foundations of their work have become more established in recent years; for example, some of the analysts view their field as a tool for presenting the objective truth. To illustrate, Morgan notes that the maturing of the risk analysis field has made it "possible to examine potential hazards in a rigorous, quantitative fashion and thus to give people . . . facts on which to base essential personal and political decisions."[1]

Others performing PRAs or PSAs propose that the probabilistic assessments are tools for measuring experts' degree of belief.[16] Of these, Watson has argued that risk analysis should provide "a rational framework for the debate about safety. . . . The argument about safety should be on the adequacy of the model, the nature of the expert judgment, the quality of the computer codes, and so on."[17] However, neither groups' claims to the rational or scientific basis of their assessments seem to have affected the public's views.

Why then does this adversarial situation exist and seem to be worsening? More basically, why do the technical experts and regulators view risk as they do and so differently than the public does? Why has informing the public about the scientific basis of risk assessments failed to change the public's views? How do feelings enter into thinking about risk? In particular, why are the public's feelings about nuclear risk so resistant to change or scientific counterargument?

To address these questions requires an understanding of how people think about risk. This article next describes the physiological mechanisms involved in thinking about risk.

PHYSIOLOGICAL, EMOTIONAL, AND COGNITIVE MECHANISMS INVOLVED IN THINKING ABOUT RISK

Physiological Mechanisms Involved in Responding to Imminent Risk

The physiological responses to danger form the foundations of our thinking about risk. Physiological mechanisms are the neurological and biochemical processes that allow us to quickly assess imminent danger and respond (e.g., fight or flight). The latest information on how these mechanisms work comes from studies of the fear response—the reactions of animals facing threatening situations (e.g., their muscles contract, they startle easily, and their blood pressure and heart rate increase).

One important finding has been that the fear response relies on crude cognitive information processing.[18] Take, for example, a hiker hearing a rustling sound and seeing a coiled slender form on the path ahead. The stimulus from the auditory system (hearing a rustling sound) or the visual system (seeing a coiled slender shape) is processed by the thalamus and passed to the amygdala as a possible danger—snake—that then causes the heart rate and blood pressure to increase and muscles to contract in less than a second. LeDoux believes that the fear response is "quick and dirty" for a reason—evolutionary adaptation. He argues that it is an evolutionary adaptation because (1) it is fast and therefore potentially life saving; and (2) failing to respond would be more costly to survival than responding inappropriately to something benign.

Another of LeDoux's findings is that the fear response results in relatively permanent emotional memories. Emotional memory is our access to the consequences (the way that we feel and the way that we behave) of an unconscious emotional process; for example, if we were in a car accident, we would remember our feeling of panic, our body tensing for the impact, and so on. It has been shown that the "amygdala (a small almond-shaped body in the center of the brain) plays an essential part in modulating the storage and strength of memories."[18] Thus fearful responses are learned quickly but are not, correspondingly, forgotten quickly. Indeed, LeDoux argues that emotional memories are not erased, but rather, fearful behaviors are controlled by the part of the brain responsible for more sophisticated information processing—the cortex.

LeDoux's findings may be important to understanding human response to risk. His work indicates that responding to danger is a physical and cognitive priority—that our history as a species may favor a quick and extreme reaction to any perceived risk. The fear response may explain, in part, why people react more strongly to the potential hazards of a new technology than to its benefits and why once fear has been aroused, it is slow to subside.

A third characteristic of the fear response is that emotional memory and the memory of "things" combine seamlessly in our conscious experiences.[18] (The learning of things is mediated by a separate system from emotional learning.) Thus we can simultaneously recall both the details, such as where and how an accident happened, and the emotional memories of how it felt. Similarly, in thinking about the details, we may reexperience the same emotions, fear, and anxiety that we had at the accident scene.

This interplay of emotional and nonemotional memories is relevant to other situations beyond the fear response; namely, everyday life. Our feelings influence our thoughts and actions;[19] and our thoughts, in turn, can determine our emotional states (e.g., imagining the worst can cause frustration).[20] How our emotions and thoughts intertwine when we consider risk will be discussed in detail in the following text.

Cognitive Aspects Involved in Anticipated Risks

The emphasis from here on will be on future risks rather than on immediate dangers because those are what people anticipate when they consider the risks posed by technological change. In general, people spend more time worrying about risks than responding to immediate threats.

Two aspects of anticipation of risk are of interest here: feeling-based thinking and intuitive modeling. Feeling-based thinking occurs when people are thinking of a feeling, such as an emotion like anxiety, or a physical feature like tightness in their gut. Thinking is involved because the individual must interpret what is felt and what this feeling means; for example, an individual might think to himself, "Is this tight feeling in my gut from hunger or uneasiness? If I'm uneasy, why am I uneasy, and what can I do about it?" (Modeling, the means by which the individual thinks about such things, will be discussed separately in the section on Characteristics of Modeling.)

Feeling-based thinking can be viewed as a more complex version of the emotional mechanisms of the fear response. As LeDoux proposes,[18] people's experiencing of feelings arises from the system that forms the basis of the fear response. With the fear response, thought processes have to be quick and dirty; with anticipating risk, however, there is time to deliberate. Thus thinking about future risk allows more complex information processing (such as involving the cortex of the brain) and interpretation. New research on memory clearly links the emotional mechanisms of the fear response to everyday living.[21] This research shows that memory is boosted by everyday emotions, such as being worried or a little scared. Our emotional memory seems to work in graduations, activated in proportion to the emotional charge.[22]

Proposed Characteristics of Feeling-Based Thinking

Feelings are used to evaluate corporal, emotional, or ethical states. People typically employ this kind of thinking to check their corporal, emotional, or ethical feelings in a projected situation; for example, a person may mentally ask himself "Would I feel good about that?" and check his body for a feeling of tightness or discomfort.[23] In this way, feeling-based thinking gives quick feedback to the individual about contemplated decisions or actions.

Feelings are likely to enter into decisions about the acceptability of risks. Because feeling-based thinking is used by individuals to assess their feelings about situations, it naturally enters into decisions about values—"what do I desire or what do I consider good?" The social acceptability of risks has recently been defined as a question of values rather than of facts.[1,5,6] Thus feeling-based thinking could play heavily in the arena of people's perceptions of the acceptability of risks.

Feelings are trusted more than reasoning in decision making. People are in the habit of using feeling-based thinking to mentally check projected situations; so they are comfortable with this process and trust it. Indeed, people often believe that their feelings provide them with a deeper truth or with a more reliable guide for decision making than reason;[23] for instance, Ann Landers advises her readers to follow their guts. "I'm a great believer in trusting one's gut. My own has never failed me. What one feels is often more important than what one thinks."[24]

Even risk analysts are not exempt from following their guts; for example, Lewis, a risk consultant for

50 years, said that he had trouble using seat belts in cars. "He would never fly a plane without fastening up, but felt that the seat belt in his car was an intrusion on his free will. He says it was perfectly normal and yet irrational."[25]

Feelings carry the same convincingness as do the emotional memories from fear responses. The primacy of feeling-based thinking partially explains why people's feelings about risk are not responsive to logic, especially others' logic.

What is felt is taken for reality. Evidence that people believe what they feel comes from studies of their perceptions. Samuelson notes that the real dangers of daily life are low and decreasing (e.g., as shown by statistics on crime and on health, safety, and environmental hazards), whereas our fears are "high and rising." He attributes this phenomena to our being inundated by psycho-facts:

> beliefs that, though not supported by hard evidence, are taken as real because their constant repetition changes the way we experience life. We feel assaulted by rising crime, increasing health hazards, falling living standards and a worsening environment. . . . The underlying conditions aren't true, but we feel they are and, therefore, they become so.[26]

One reason why psycho-facts affect our feelings and subsequent perceptions of reality is emotional memory. As Cahill's research showed,[21] we selectively remember the news that upsets us. This means that the public is likely to forget the neutral information on a much publicized technology, like nuclear power, and to remember only the news that worried them. Over time, it is possible that some people may develop an aversion to a technology, where just thinking about it makes them feel revulsion. This revulsion toward nuclear technologies has been documented among many populations, as was described earlier in the section on nuclear images.

Feelings play a special role in alerting people to risk and letting them know when they have dealt with it. Although people's dealings with risk involve visual, aural,[a] and feeling-based thinking, the feelings are likely to be key. In particular, feelings are the means by which individuals become aware of risk. This process will be described in detail in the section on Interaction Between Feelings and Modeling.

[a]In addition to thinking by means of feelings, people can mentally think in terms of pictures or sounds.[27] Each of these ways of thinking has its own flavor. Visual thinking has the nuance of viewing something from a distance, such as when one replays where one has been to find a lost object. Aural thinking has the flavor of mentally monitoring one's place in a process (e.g., "I've done this and this and need to do that next") or of cautioning ("this situation could backfire") or criticizing oneself and others.

Characteristics of Modeling

In addition to feeling-based thinking, individuals use mental models in dealing with risk. Models are defined as "selective abstractions that help users identify, explain, predict, and control events in the world."[28] Given this definition, worry is one form of modeling;[b] for example, when we worry, we

> create scenarios or images of impending events based in part upon what we feel is fairly certain in our future and in part on vague notions of what we believe is possible (rather than probable). Such scenarios could serve as means for understanding ways in which future events might be realized and could be useful in preparing to meet them.[29]

Models are created in interaction with the situation. The models that the individual creates are done so in interaction with the situation.[30] The situation broadly includes the things in time and the environment that are associated with the risk—for instance, the person's concept of self (e.g., what he thinks he can or should do), past experiences (e.g., factual and emotional memories), and other cultural factors.

Note that individuals may create more than one model per situation, and these models may be logically inconsistent. Take, for example, someone who is worrying. The individual "may create a spectrum of potential scenarios, some of which may be mutually exclusive, and proceed to worry simultaneously about the outcome of each. . . ."[31]

Models include assumptions about how things function. Individuals' models include implicit assumptions about how things in the world function or are related, especially causally. Individuals use their understanding to try to control outcomes, in this case, to avoid the negative consequences of risk.

Take, for example, the area of risk to health. In one woman's model, becoming chilled was the cause of head colds. She stated that she and two members of her family came down with bad head colds this summer because they had become chilled by the extreme air

[b]Note that the running of computer codes for assessing technological risks and this report's description of human thinking about risk can be viewed as other forms of modeling; they are external manifestations of the human capacity to mentally model.

conditioning in church. Her implicit model was that exposure to cold temperatures, not germs, caused the head cold. At a more detailed personal level, she believed that head colds caused her to have subsequent earaches, chest colds, and laryngitis.

Models imply a course of action. Individuals employ their models in thinking about how to achieve something desired or to avoid an unwanted consequence. Thus, in the preceding example, the woman's model led her to dress warmly year round to avoid getting a cold.

Models behave like reality in the expectations of their users. Compton et al. point out that a "basic feature of a model is that it can simulate reality, behave like reality according to the expectations of the users. . . . The model is not the thing but it behaves like the thing, not in an absolute sense but according to the expectations of the. . . users;"[32] for instance, in the head cold example, the woman acts as if her model were reality. She follows the implied course of action in the belief that it will save her from getting a bad ear infection. She does not consider her thinking as a model, an abstraction, or a simplification of reality.

Models are often based on illusory correlation. The assumptions in the individuals' models are often wrong because humans and other animals frequently assume that things covary, when, in fact, they may not.[33,34] This tendency is called illusory correlation. Illusory correlation stems from a very basic tendency—to mentally associate events that occur together in time.

This process of linking events has allowed creatures through time to learn causes and effects and to manipulate them; for example, pigeons will associate random feedings with whatever motion (e.g., hopping) they were making just before the food arrived. Baseball pitchers will associate the onset of a losing streak with some unrelated object, such as an unlucky hat, and will subsequently avoid it as taboo.[35] The tendency to link events and assume causal relationships is the basis for animal training, fear conditioning, and the creation of rituals or taboos;[35] however, this process often leads to incorrect assumptions about how things are related. Thus, in modeling, the individual is often mistaken about what causes a particular loss and how to avoid it in the future.

Models are updated. The person's models change in interaction with situations; for example, the woman who believed that cold temperatures caused her to get head colds had learned this from her mother. Her mother had provided her with a background of what caused head colds—exposure to cold. Her mother also said that going from extreme hot to cold temperatures caused bladder infections; however, the daughter changed her submodel as a result of a situational experience: "I learned my mother was wrong when I was in Florida because we went from very hot to very cold in the Pizza Hut. The cold made my bones ache, like they feel before you get the flu, but I never got a bladder infection."

For models to evolve, they need to be replaced by better models (that is, models that are less wrong). Models are described as less wrong rather than right because all models are simplifications and therefore, even at best, cannot be correct.[a] Thus, for models to evolve, they must be recognized as wrong and replaced with something less wrong.

Models do not always evolve, however. Sometimes they are replaced by models that are more wrong, such as when a person overgeneralizes from some traumatic experience (e.g., when a person concludes that cars cause death). More commonly, though, people's models are slow to change; for example, it has been noted that people's illusory correlations "can persist in the face of disconfirming evidence."[33]

Interaction Between Feelings and Modeling

This section illustrates how modeling and feeling-based thinking might interact in situations involving risk.

Feelings alert individuals to the presence of risk. Feeling-based thinking is the cue that alerts individuals to an anticipated risk; for example, in thinking about an upcoming situation, such as a tax audit or voluntary surgery, the person's emotion could be fear, anxiety, worry, or dread; his feelings could be vague uneasiness or physical symptoms, like a queasy stomach, tight chest, racing heart, or sweaty palms. Generally, the feelings that alert the individual to risk will be those judged unpleasant or negative.

After the negative feelings have alerted the individual to risk, the individual begins modeling the situation. After the negative feelings have alerted the individual to the possibility of loss, the individual starts thinking about the situation, modeling it as described

[a]In addition, models cannot be proven correct, only wrong. According to the philosopher Popper,[30] models are like hypotheses, and hypotheses can never be proven true—only false.

in the preceding section. Modeling is part of the individual's response to the risk; it allows the individual to relate aspects of the situation to the outcome. Generally, individuals will try to change either the anticipated undesirable consequences, their negative feelings, or both.

One means of changing the consequences is to take action to limit them. The options for action are typically implied by the individual's model; for instance, the woman in the earlier example was alerted to the risk of an ear infection by her cold symptoms. Her model predicted that colds lead to ear infections unless she kept her ears free of fluids. Thus she took action—decongestants—to keep her colds from spreading to her ears.

One means of changing the negative feeling is to deny it; for instance, the woman could have denied that she had a cold, thinking, for example, "I don't have a cold because I can't afford to be sick now. Maybe it's just allergies." Other means of changing the negative feeling are by taking drugs, exercising, praying, meditating, or engaging in relaxation techniques. Note that the individual can employ a variety of strategies in changing the consequences or the negative feelings and that these are likely to overlap.

Feelings are the means by which individuals judge that they have resolved the risk. Feeling-based thinking is the means by which individuals know that they have resolved the imagined or actual situation, that their efforts have worked, and that they can go on to something else. The kinesthetic feedback that allows them to go on is typically a positive feeling of confidence, an "it's-going-to-be-all-right mood," a relaxed state, or an "at-peace" sensation. Individuals may check their feelings as a conscious judgment, such as when they decide that they will keep thinking of solutions until they find one with which they are comfortable, or people may be unaware that they are using their feelings in this manner until it is called to their attention.

If people do not receive positive feelings as feedback, they may become stuck in the unpleasant state of uneasiness, or worse, fear. In the extreme case, this most often occurs after the individual has been a victim of some traumatic experience. No matter what coping strategies the individual employs, his mental check just reveals muscular tension and fear. Owing to the relative permanence of emotional memories from a fear response, this situation may continue for a long time.

For example, a woman who had been in a car accident reduced her amount of driving and avoided busy thoroughfares; however, these strategies did not help greatly:

> When a "car follows too closely, her heart races. She locks her jaw and tenses her muscles." . . . She has flashbacks to the accident in April that totaled her Mitsubishi Mirage. She's afraid to drive. "I feel like that impact is going to come again," . . . . "I just want to get into a car and drive without worrying."[36]

More commonly, however, individuals become stuck in receiving slightly negative feelings as feedback over a shorter period of time. This often occurs when they are trying to make a difficult decision and none of the alternatives are totally satisfactory. Negative consequences or risks are associated with each option. The individual knows that he is not happy with the status quo (this is the negative feedback) and thinks of alternatives but is not comfortable with these either (this is also negative feedback).

CULTURAL FACTORS AFFECTING APPROACHES TO RISK

Cultural factors condition how people perceive and react to risk. Culture is simply defined as what people learn socially as members of groups. Groups include family, religious, educational, vocational, and interest groups. People learn beliefs, values (e.g., what is good), and norms for behavior (when is it appropriate to act a certain way). The learning can be conscious, such as through receiving instruction, or unconscious, such as by emulating others. Through this process, people internalize the group's culture; that is, they adopt many of its beliefs, values, or norms as their own. As a result, they are likely to view things from their culture's perspective and to perceive its ways as superior.[37] This tendency—ethnocentrism—operates in the risk arena and accentuates the differences between groups.

People Learn Mental Models, Emotions, and What Is Considered Risky from Their Cultures

People learn models. Individuals learn their culture's beliefs on how objects in the world are related. These beliefs are, in essence, models; for instance, members of western scientific culture believe that colds are caused by germs and that these are transmitted to others via their nose, eyes, or mouth. Thus members of this culture believe that sharing silverware with a cold sufferer is risky behavior.

These models are viewed as indisputable by the members of a culture; for example, just as many in the Western scientific world consider germs to be the root explanation of illness, many in Africa (even those trained in Western medicine[38]) believe witchcraft to be the underlying cause. To each, his own model is truth; the other's model is wrong, naive, or nonsensical. This kind of thinking illustrates the effect of ethnocentrism.

People learn when and how to feel and express emotions. Individuals learn what is valued, and this sets a context for the experiencing of emotions; for example, members of Western society tend to value sanitary conditions as *good,* and given their belief in the germ model, *healthy.* Thus members of this culture are likely to feel revulsion at the thought of flies climbing on their food or faces.

Individuals also learn the context in which an emotion is felt and expressed; for example, what makes a person angry depends upon those situations or events which are considered by his culture offensive or frustrating.[39] For instance, the Mescalero Apache Tribal President has proposed to open a monitored retrieval storage facility for spent nuclear fuel rods on the reservation in New Mexico.[40] Angry antinuclear activists have chanted protests outside the reservation. These protesters were not members of the tribe. Tribal members may not have felt angered by their tribal leader's proposal, or they may not have considered it appropriate to express their anger overtly.[a]

Note that individuals are sometimes trapped in feeling their own emotions. They may wish to feel otherwise but be unable to do so because they have learned emotions in a single-minded way;[34] for example, they may have learned that snakes are frightening or that "no nukes are good nukes." They have become victims of their own mental associations.

[a]In January, Mescalero Apache tribal members voted 490 to 362 against the creation of the storage facility. The tribe's manager for the project attributed the loss to "fear of possible contamination by the fuel rods and to ignorance about the project."[41] Six weeks later, the tribe voted in favor of the storage facility.[42]

Scientists Belong to a Separate Subculture and Are Socialized Differently Regarding Risk

As a result of their educations and on-the-job training, scientists can be considered as having been socialized in a separate culture.[b] This separate culture takes a different approach to risk than the public does. Scientists, as a whole, have been shown to perceive less nuclear-related risk than the public;[43] for example, scientists believe that nuclear wastes are less risky than most lay persons believe they are.

In addition, subcultures within the scientific culture itself hold different views of risk. Barke and Jenkins-Smith[43] have found that scientists' views of nuclear risk correlate to their field (e.g., physics, biology, and engineering) and to their type of employer (e.g., Federal Government, local government, and academia). For instance, scientists working for the Federal Government judge the risks posed by nuclear waste to be less severe than do scientists in local governments and academia, and physical scientists perceive nuclear risks to be lower than do life scientists. In addition, scientists working for the Federal Government consider environmental restrictions to be a less valid means for dealing with nuclear risks than do those employed by universities or local government. In sum, scientists who work for local governments and academia have views of risk closest to those of the general public; those working for the Federal Government hold the views of risk furthest from those of the public. Thus NRC scientists, with their physical science background and federal employment, are likely to view risk very differently than the public.

The work that scientists perform can further condition their views of risk; for example, work that involves modeling, such as of reactor accidents, is likely to make its practitioners prone to a particular cognitive bias—*illusion of control.* Illusion of control occurs

[b]Vocational selection may also be a factor in explaining why scientists differ from other groups. Research on vocational selection indicates that people are drawn to fields and jobs that match their interests and preferences; for example, according to a well-respected personality diagnostic (Myer-Briggs), the majority of people in the United States are extroverts, whereas the majority of scientists are introverts. This is an important difference because introverts tend to focus on the inner world of ideas, and extroverts, on the outer world of people. Thus scientists may be predisposed by their very personalities to approach risk differently—more conceptually and less personally—than the public.

when people acquire the impression of having more control over outcomes than is justified. Illusion of control has been shown to occur in individuals when they have spent time analyzing a situation or observing a sequence of successful outcomes.[33,34] Thus those who go through the thought processes intrinsic to modeling are particularly prone to this cognitive bias. This bias could cause them to unconsciously underestimate the magnitude of uncertainty present in the model.[44] As a result, they could tend to give more credence to the model's predictions than is warranted.

SUMMARY

This work was initiated to explore some questions about how people think about risk. In this section, the findings will be summarized by questions.

• **Why do technical experts and regulators view risk so differently than the public?** Cultural conditioning is largely responsible for the different approaches to risk. Technical experts and regulators are mostly scientists and as such have been socialized by their educations, work, peers, and organizations to view risk differently than the public.

• **Why has informing the public of risk findings—the approach used throughout the government—had so little effect? How do feelings enter into thinking about risk? In particular, why are the public's feelings about nuclear risk so resistant to change or scientific counterargument?** Informing the public about the results of science-based risk analyses and assessments does not change their thinking for several reasons. First, for people to accept the scientific views of risk, they must view "science," or at least rationality, as a higher authority. They would be more likely to do so if they had received the same cultural conditioning as scientists. Because they have not, the public is likely to have their own ideas of higher authority and question the pronouncements of science in their personal lives.[6,45] Then, too, the public's views of risk are affected, usually negatively, by their confidence in those managing the technology.[11,12] Second, much of the public's thinking about nuclear risk involves averse feelings, and these feelings have been identified as resistant to change. Averse feelings may be long lasting because of the way in which emotions are learned or because of their link to a survival mechanism—the fear response.

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Twenty-Third Water Reactor Safety Information Meeting

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Abstract: *This article is a brief review of the* Twenty-Third Water Reactor Safety Information Meeting, *one of an ongoing series of meetings sponsored by the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research (RES). The purpose of these meetings is to give the nuclear safety community up-to-date information on NRC-sponsored and other research in the nuclear-power-reactor safety field. A table listing all the papers and authors is included, and highlights of the meeting are provided.*

More than 300 people, representing 23 countries, attended the *Twenty-Third Water Reactor Safety Information Meeting* (23WRSIM), held at the Bethesda Marriott Hotel in Bethesda, Maryland, on October 23–25, 1995. Papers from France, Italy, Norway, Russia, Sweden, and Switzerland, as well as the United States, were presented. Every important aspect of water reactor safety was covered. The presentations also addressed the next generation of reactors and concerns regarding operating reactors.

Twelve technical sessions, including two on severe accident research, were held. Table 1 lists the 63 talks and papers (including the plenary session) presented at the meeting. Summaries of the presentations are included in the transactions,[1] and the complete papers are published in the proceedings.[2]

The meeting is one of a series previously known as the Light-Water Reactor Safety Research Information Meetings. The elimination of the words *Light* and *Research* was intended to convey a somewhat broader coverage, one that would also include heavy-water reactors and safety matters not as strictly tied to research activities as was formerly the case.

In this article, selected papers and talks are briefly summarized. Space limitations prevent covering more than just a few papers. The inclusion of a paper is not meant to indicate preferences or importance. Inclusion does, however, mean the paper addresses what are considered to be the more timely issues facing the industry.

PLENARY SESSION AND LUNCHEON SPEECH

The keynote address for the meeting was given by the Chairman of the Nuclear Regulatory Commission (NRC), Dr. Shirley Ann Jackson, who pointed out her understanding of ". . . the role that research plays in the fulfillment of agency and corporate missions." Dr. Jackson then stressed how NRC research programs furnish the technical basis for regulatory programs, which, in turn, ensure that public safety is not compromised: "Clearly, valid regulatory decisions must be based on the firm technical understanding that comes from research. NRC research programs provide a strong independent technical capability for our regulatory programs. Without this strong technical component, our decision-making capability would be diminished and public safety could be compromised." Dr. Jackson addressed two general areas regarding NRC research: (1) industry use of NRC research and (2) the future of the NRC research program. Excerpts from her remarks follow:[3]

INDUSTRY USE OF NRC RESEARCH

The research programs have provided significant enhancements to data bases of all kinds, and they have produced analytical methods and measurement techniques that have been very useful to the nuclear industry.

Behavior of Emergency Core Cooling Systems

In the early 1970s, NRC conducted a major hearing on the behavior of emergency core cooling systems (ECCS) during loss-of-coolant accidents. As a result, NRC undertook a number of research programs to confirm judgments that were made in the hearings. Without the promise of that comprehensive research program, it is quite possible that no construction permits or operating licenses would have been issued in the mid 1970s. In that

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decade, our research confirmed the conservatism in oxidation kinetics, decay heat levels, embrittlement criteria and other requirements. Arguments over ECCS analyses no longer held up licensing activities.

Heavy Section Steel Research

In another area, where work was started early by our predecessor—the Atomic Energy Commission—the heavy-section steel program provided essential information on materials properties and fracture behavior of steels and weld materials. Virtually all of the early research on the effects of irradiation on the embrittlement of reactor pressure vessels was performed in NRC research programs.

Piping Integrity Research

NRC's piping integrity research provided much of the basis for relaxing earlier requirements and allowing leak-before-break to be considered.

Risk Assessment

One of NRC's most important research accomplishments is in the area of risk assessment. It is fair to say that the NRC's research program has had a major impact on the discipline of probabilistic risk analysis and particularly in its application to nuclear reactor safety. While elements of risk assessment had some earlier use in the aerospace industry, the NRC's Reactor Safety Study (WASH-1400) represented the first integrated assessment of nuclear plant risk ever done. Our later assessment of severe accident risks (NUREG-1150) provided better estimates of plant risk based on a more complete understanding of severe accident phenomena. All U.S. nuclear plants have now performed risk assessments and, because of this work, the increased use of risk insights in regulatory activities holds the potential to improve safety and at the same time reduce costs.

Severe Accident Research

WASH-1400 demonstrated that essentially all of the risk from nuclear plants comes from core-melt accidents. Therefore, right after the accident at Three Mile Island, the NRC initiated an aggressive severe accident research program to examine core-melt sequences and subsequent accident phenomena that might challenge containment integrity. The understanding of severe accident phenomena from this research allowed the risk assessment methodology of WASH-1400 to be revisited to produce the credible assessments of NUREG-1150. Although, it may not be generally realized, more than 75 percent of all the severe accident research done in the U.S. has been done by the NRC. The Department of Energy work on the examination of Three Mile Island represents about 10 percent of the severe accident research in the U.S., and a little less than 15 percent was done collectively by the industry under EPRI [Electric Power Research Institute] and the Industry Degraded Core Rulemaking program. This large body of severe accident research has been used by all U.S. utilities as the basis for their risk assessments and Individual Plant Examinations.

Source Term Research

Revised source terms, which have come out of our research program, include a more realistic timing and fission product composition and will provide the basis for the design and operation of plant features to mitigate fission product releases more readily.

Current Research Activities

The NRC is also working in newer areas. Let me identify some of these areas, and you will hear more about them in the technical sessions that will start later today.

Most of you are aware of a recent concern about high-burnup fuel and our current program in that area. We issued an Information Notice to the utilities last year on this subject, and it was discussed last year at this Water Reactor Safety Information Meeting. Using data largely from foreign sources and a strong domestic analytical effort, we are hoping to revise fuel damage licensing criteria without creating undue penalties for the operating utilities.

Our progress on the thermal hydraulics of the advanced pressurized water reactor, the AP-600, has been significant, including scaled testing in the ROSA loop in Japan and in a test loop at Oregon State University, and detailed studies utilizing the RELAP computer code are being performed. Several design changes in the AP-600 have been made by Westinghouse that can be attributed, at least in part, to results from these programs.

Hardened vents in Mark-I containments and provisions for flooding the Mark-I basemat to prevent liner attack are examples of actions taken as a result of our core damage research. Of no less importance, the severe accident research program has provided the basis for not doing anything in other areas. For example, early consideration of the possibility of direct containment heating in PWR containments suggested a high likelihood of early containment failure. More detailed and structured results from this program have indicated that direct containment heating is a very small contributor of public risks at most plants.

Present work on piping integrity is almost complete for the large baseline programs of the International Piping Integrity Research Group. We are now pursuing more specialized studies. For example, in collaboration with the Nuclear Engineering Power Corporation (NUPEC) of Japan, we are participating in large-scale seismic tests of main steam and feedwater piping systems on the largest shaking table in the world.

International Safety Research Cooperation

After the accidents at Three Mile Island and at Chernobyl, nuclear safety has been increasingly recognized as a world-wide concern, and our cooperative programs have intensified. This trend has been further enhanced by the general reduction in research budgets at home and abroad, and the resulting need to pool resources in cooperative programs.

Some of our cooperative programs, like the Halden Project's fuel behavior work and the International Piping Integrity Review Group, have continued over many years, and we are still participating in their activities. Others, like the high-burnup fuel tests in the French CABRI reactor and the containment integrity program with NUPEC in Japan, are relatively new cooperative programs.

THE FUTURE OF THE NRC RESEARCH PROGRAM

We will continue to work in important areas like thermal-hydraulics, materials, severe accidents, and risk assessment; and we will continue to participate in, and to stay abreast of, international nuclear research programs.

We are now in a period of change at the NRC. The electric utility industry is under strong competitive pressures and is diligently examining means to reduce its costs. NRC has a role to play in reducing the regulatory burden when the safety benefit is marginal.

However, even without external pressure to reduce costs, a new culture, which I refer to as risk-informed, performance-based regulation, is being adopted by the NRC. NRC is becoming less prescriptive and more performance-oriented in its regulatory posture in order to provide greater flexibility to licensees while maintaining adequate protection for the public. Cost-consciousness and cost-effectiveness pervade all of NRC's operations, including research.

NRC's research programs are being reexamined to ensure proper focus under this new paradigm. Research planning must consider the current and prospective level of plant safety, and there should be a reasonable expectation that research projects and their results will be cost beneficial. Among the criteria to evaluate the merits of a research project is the likelihood that the results will improve the effectiveness of regulations and minimize any undue burdens they impose. Some of the rules that the NRC developed conservatively in the 1960s and 1970s because of lack of information can now be modified as a result of improved knowledge that has been gained through investments in research over the past 20 years. Future investments in research will be expected to continue this trend.

As nuclear power plants age, we must examine the standards and operating procedures that have been imposed on critical components, such as the primary coolant system boundary, to assure ourselves and the public that an adequate safety margin still exists. Only through research can we derive dependable estimates against which we can make such judgements. One of our top research priorities is improving our understanding of the aging processes in nuclear power plants with particular focus at the present time on reactor vessels, steam generators, and electrical cables.

Many of the performance standards will be established by drawing on knowledge developed in risk assessments performed both by NRC as well as licensees.

However, we must acknowledge realities. Careful evaluation is needed to determine the future value of additional research in all areas. We are approaching the point where we can, in some areas, go into a program maintenance mode that includes a very limited experimental program and thoughtful fine-tuning of existing analytical models. Our current international cooperative experimental programs are expected to provide additional data to help make this determination. In doing this, however, we have found that adequate resources and careful planning are still required to avoid losing the important technical skills.

However, further emphasis and new work is needed in important areas related to changing focus of our mission, i.e., risk assessment research to develop and strengthen methodologies for dealing with human/organizational factors and degraded equipment. New methodologies from other fields need to be developed and applied to age related effects in reactors, i.e., going beyond fracture mechanics to relate detailed measurements using new experimental probes to microscopic materials properties in order to make stronger predictive statements about behavior, as well as development of possible in situ probes of key plant systems such as the Reactor Pressure Vessel itself.

I know that countries already share the results of their reactor research and that in some specific technical areas, a number of countries have joined together to address issues of common concern and interest. We need to be certain that our collaborative research projects recognize and build upon the unique areas of specialization and particular expertise each of us has. Through existing institutions, such as the Committee on the Safety of Nuclear Installations of the OECD [Organization for Economic Cooperation and Development] Nuclear Energy Agency, we must more diligently focus our attention to the planning and integration of our research efforts. At the same time, we should hold these institutions to high performance and efficiency standards so that value is achieved from our investments in them. But, I would like to propose that we go much further. I think that we should consider an international reactor research program focused on aging and risk assessment methodologies in which we seek to integrate the regulatory research activities of various countries within the context of a formal international research program. Each country could specialize in areas of its particular expertise. Thus, we would avoid duplication of effort and meet the common challenges which we are encountering and the common downward pressures on our various regulatory research budgets.

Strategic Assessment and Rebaselining

We still need to develop a strategic vision that allows us to adapt to a changing environment and to budgetary constraints, to carry out our regulatory programs more effectively, to take on possible new missions, to conduct effective resource planning, while remaining responsive to the public and the regulated industries. The first phase of the strategic initiative, the "strategic assessment," involves identifying and examining the sources of the mandates that make up our regulatory mission—statutes, executive branch directives, and Commission decisions—so that we can establish a mutual understanding of what the NRC mission is and what is required of us. Also included in this phase is a process of looking at all agency activities to determine whether they are being conducted in response to a specific mandate or whether these activities have some other rationale for their existence, and whether there are areas where we should have ongoing programs to implement specific missions, but do not. This phase is also meant to begin to surface key strategic issues, questions, and decision making points, which the Commission will address.

The subsequent phases—rebaselining and strategic planning—flowing from Commission decisions on the key strategic issues, questions and policy alternatives, will address what our programmatic needs are and should be, and what resource levels should be assigned to them. The first phase will drive and provide input to the following phases and ultimately to budget and human resource planning, which is the final phase. I believe that this review is necessary to position us to meet effectively the challenges we face and to guide intelligently our activities and decision-making in the future.

Following Dr. Jackson's remarks, William T. Russell, Director of NRC's Office of Nuclear Reactor Regulation (NRR), presented current issues in nuclear reactor regulation as they relate to research activities. Mr. Russell's remarks, in part, were as follows:[4]

So, let me start off by saying that I have not tried to give a comprehensive listing of issues that are currently facing the reactor program, but rather to select those that I thought were relevant as they relate to research activities. Use of probabilistic risk assessment in regulatory decisions; materials aging issues concerning steam generators and reactor vessels; high burnup fuels; accident management; and digital instrumentation and control are just a sampling of the important issues that I want to talk about.

Probabilistic Risk Assessment (PRA)

The Commission in August passed a major milestone with the publication of a policy statement on the use of probabilistic safety assessment in regulation. We had been for some years using probabilistic safety assessment techniques starting with WASH 1400. Shortly after Three Mile Island, the "Lewis" report reviewed the use of PRA in regulation.

In the intervening years, from 1980 to 1995, there has been substantial progress made in the use of PRA, and this is recognized in the Commission's policy statement, which encourages, to the extent supported by data and by the state-of-the-art, the use of PRA in expanding regulatory activities.

A precedent was set when four NRC office directors got together and developed an overall plan for how the agency should proceed. These four offices were NMSS [Office of Nuclear Material Safety and Safeguards], RES, NRR, and AEOD [Office for Analysis and Evaluation of Operational Data]. We proposed that there be an agency-wide action plan for coordinating these activities. Within that plan, there are a few areas that should be highlighted that related to the future use of PRA and our interface with the NRC research program.

One area in particular deals with in-service testing requirements. These requirements flow from Section XI of the ASME [American Society of Mechanical Engineers] Code.

Another area that NRR is working on very hard and that we have had feedback on is implementation of the new maintenance rule.

An important part of implementation of the maintenance rule is the use of risk insights on a plant-specific basis to develop the list of those systems, trains, and selected components that have a relatively high risk significance and to monitor and trend their performance.

There is a major effort being looked at by the RES to try to develop appropriate standards for a PRA when it is being used in a regulatory context. RES and NRR are respectively working on the development of a Regulatory Guide and Standard Review Plan for various applications of PRA to regulated activities.

Steam Generators

In the aging area, one aging problem—and a real current problem—is the issue associated with steam generators. Proceeding essentially on a case-by-case basis with what I call crisis management, we continue to identify degradation mechanisms in steam generators. This fall has been a particularly heavy time, with the identification of additional circumferential cracking in Combustion Engineering generators, and much more substantial axial cracking in the area of the support plates in Westinghouse generators. In many cases, licensees have not anticipated the degradation sufficiently.

Our objective for the rulemaking is that it be performance based. That is, we would establish objective criteria, probably along the line of Regulatory Guide 1.121 as it relates to structural integrity margins to be demonstrated.

We have three different issues to examine: (1) normal operation with respect to primary-to-secondary leakage through generators, which is relatively easy to detect with on-line monitoring for radioactive materials leaking through the generators; (2) in the context of accident analysis in the past we have used very conservative approaches, looking at, essentially, one gallon per minute primary-to-secondary leakage through the generator with very conservative calculations for iodine spiking factors and atmospheric dispersion; and (3) severe accidents, particularly for transients resulting from loss of secondary heat sink where you may be either boiling off the primary, ending up with hot gases, or conditions under which you might have high-pressure core melt or core damage sequence.

The implications of a degraded generator and what it means as it relates to a potential containment bypass scenario are significant and need to be examined in an integrated way. We hope to have a three-tiered approach. That is, the performance-based rule would establish the framework and the objectives. Next, we would expect to have a regulatory guide to identify various methodologies for qualifying, for example, inspection techniques and providing guidance on the type of analysis to be done. Below that we would have a number of topical reports for various degradation techniques or for various vendors for the types of inspection techniques that they may use. These reports would be reviewed and compared to the guidelines in the regulatory guide.

Reactor Pressure Vessel

The next issue is a critical one: reactor pressure vessels. The Commission has rulemaking under way to address procedures for annealing. You are aware of the pressurized-thermal-shock (PTS) rule and some of the other requirements of Appendix G, fracture toughness, and of Appendix H, surveillance programs.

We published a NUREG report containing the data that have been received by NRC in response to the earlier Generic Letter 92-01. We hope the industry will take this to the next step and include all data, and that we will resolve the issues with respect to treatment of data as proprietary. Most of you are aware that I have sent out letters to deny withholding of proprietary information on the basis that the information constitutes information important to reactor safety and therefore should be available to the public so the bases for our conclusions on such issues of vessel toughness and the ability to meet the pressurized-thermal-shock rule can be understood.

High Burnup Fuel

High burnup fuel is an example of what I think is a success, both on an international level and on a research level. We learned about some testing that had been done in France and also in Japan on transient reactivity behavior of fuel under conditions of high burnup.

In reviewing the acceptability of higher burnups, we have to be concerned with both the transient reactivity behavior and also the recent experience with the Three Mile Island core. At TMI-1, failures of cladding on first-cycle fuel in a transition core occurred. The combination of high boron/boric acid chemistry along with relatively high heat fluxes caused failures of fuel pins. These issues also relate to the integrity of cladding, which, if this is occurring on first-cycle fuel rather than high-burnup fuel, concerns us.

Accident Management

Accident management is an issue on which the staff has been patient. We need to mine the severe accident research work, some of the Individual Plant Examination work, and the NUREG-1150 work on severe accidents to try to identify appropriate strategies that can be used in the context of accident management and also to develop aids to assist decision makers, and to come up with a framework for what I characterize as the Emergency Operations Facility inward aspects of responding to an emergency.

Digital Instrumentation and Control

In the area of digital instrumentation and control (I&C), I am focusing on a rather narrow aspect. The National Academy of Sciences is looking at the overall approach to instrumentation and control, particularly digital I&C, and the human factors aspects of maintaining software-based protection systems or safety systems.

James M. Taylor, NRC's Executive Director for Operations, gave the luncheon speech on October 25, 1995, the last day of the conference. Mr. Taylor addressed three areas in which reactor safety research has a vital function. Portions of his remarks are presented here:[5]

It remains clear that nuclear safety research continues to play a vital role in our agency's approach to accomplish its regulatory mission. For research not only may uncover potential problem areas, as is the case with high burnup fuel, but also offers the potential to provide the solution to such problems, as with reactor pressure vessel annealing.

I want to briefly talk to you this afternoon on three topics where reactor safety research has played, and continues to play, an important role. These areas are the increased use of probabilistic risk assessment methods in regulatory activities, thyroid cancer studies arising from the Chernobyl accident, and recent developments in reactor pressure vessel annealing.

PROBABILISTIC RISK ASSESSMENT METHODS

On August 6, 1995, the NRC published its Final Policy Statement on the Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities.

This policy statement represents the logical conclusion to a long and successful application of probabilistic risk assessment, or PRA, methods in nuclear safety research and analysis. In this final policy statement, the

Commission stated that an overall policy on the use of PRA methods in nuclear regulatory activities should be established so that the many potential applications of PRA could be implemented in a consistent and predictable manner and would promote regulatory stability and efficiency. I'd like to briefly remind you of the background and events that led to this policy and touch on the key points of the policy itself.

PRA Background

The NRC has generally regulated nuclear power plants and the use of nuclear material based on a deterministic approach. In this approach, a set of challenges to safety is determined, and an acceptable level of mitigation is defined. This leads to the so-called design basis accident, or DBA, approach.

An example is the postulated loss-of-coolant-accident, or LOCA. Here, the safety challenge is a postulated coolant pipe rupture, up to and including the largest coolant pipe in the reactor coolant system. The level of mitigation required is that the fuel cladding temperature not exceed a specified temperature in the face of this challenge. Another aspect of the LOCA postulates the occurrence of a fission product release within containment as the safety challenge, and requires that the plant design and site characteristics be such that a hypothetical individual located at the exclusion area boundary would not receive a radiation dose in excess of prescribed limits.

In 1975, the Reactor Safety Study, or WASH-1400, was published. This was the first systematic assessment of reactor risk that used modern probabilistic methods, and this report represents a seminal event in reactor risk assessment. More recently, we have seen the issuance of NUREG-1150 in 1990, which used improved PRA techniques to assess the risks associated with five U.S. nuclear power plants. This study has been noted as a significant turning point in the use of risk-based concepts in the regulatory process.

Recent PRA Applications

PRA methods have been successfully applied in a number of regulatory activities, acting as a complement to the traditional deterministic process. A number of recent Commission Policies or rules have been based, in part, on PRA methods. It may be appropriate to mention just a few of these.

These include the Backfit Rule, 10 CFR 50.109, the Commission's Safety Goal Policy, the Commission's Severe Accident Policy, and the Commission's Policy on Technical Specification Improvement. PRA methods have also been used in developing regulations regarding anticipated transients without scram (ATWS), the station blackout rulemaking, and in support of generic issue prioritization and resolution.

The NRC is also currently using PRA techniques to assess the safety importance of operating reactor events and as a part of the design certification review for advanced reactor designs. As many of you are aware, the Individual Plant Examination (IPE) Program, which resulted from the Commission's Severe Accident Policy, has resulted in power reactor licensees using risk assessment methods to identify potential plant vulnerabilities.

PRA Policy Statement

I will briefly summarize the major points covered in this policy statement. First, the Commission has stated that the use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

Second, the Policy states that PRA and associated analyses should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guides, license commitments, and staff practices.

Third, the Policy states that PRA evaluations in support of regulatory decisions should be as realistic as practicable, and appropriate supporting data should be publicly available for review.

Finally, the Policy concludes that the Commission's safety goals for nuclear power plants and subsidiary numerical objectives are to be used with appropriate consideration of uncertainties in making regulatory judgments on the need for proposing and backfitting new generic requirements on nuclear power plant licensees.

STUDY OF CHILDHOOD THYROID CANCER FROM THE CHERNOBYL NUCLEAR ACCIDENT

Another area that could potentially have significant benefit to the industry has been the U.S. Government efforts in conducting epidemiologic studies of radiation induced thyroid disease in Belarus and Ukraine.

These studies focus on the incidence of thyroid disease, especially cancer, resulting from the 1986 accident at the Chernobyl nuclear power plant. Two scientific protocols have been signed. The first was signed with Belarus on May 26, 1994, and recently, the second, with Ukraine on May 10, 1995, during the president's visit to Kiev. The purpose of these thyroid studies is to assess the risk of thyroid cancer and hypothyroidism among persons, particularly children, who were exposed to iodine radioisotopes, especially I-131, during and/or following the Chernobyl accident.

The release of radioiodine is likely to figure prominently in any nuclear power plant accident.

These studies originated under the auspices of a 1988 memorandum of cooperation between the United States and the former Soviet Union concerning civilian nuclear reactor safety following the Chernobyl accident.

Currently the studies are being primarily implemented by the National Cancer Institute [NCI] with support from the Department of Energy [DOE] and the Nuclear Regulatory Commission in cooperation with the Ministries of Health of Belarus and Ukraine and several scientific institutes in these countries.

Presently, NRC, DOE, and NCI continue to provide all the financial support for the implementation of the studies. Considering present U.S. Government budget issues, international participation may be welcome in the future.

ANNEALING OF NUCLEAR REACTOR PRESSURE VESSELS

Assuring the structural integrity of the reactor pressure vessel is fundamental to the safe operation of nuclear power plants. It has been long recognized that neutrons escaping from the reactor core can embrittle the pressure vessel materials, and, as plants age, can limit the safe operating life of a reactor pressure vessel.

Today, thermal annealing of the reactor pressure vessel is the only known technique for mitigating neutron irradiation embrittlement.

Thermal annealing consists of heating the pressure vessel beltline materials to temperatures of about 850 to 900°F, well above the normal operating temperature of the pressure vessel, and holding them there for an extended period, typically about one week. The annealing temperature must be high enough to allow atomic diffusion to take place and restore the embrittlement damage but not so high as to cause geometric distortion of the vessel or affect the original heat treatment of the vessel materials.

Reactor pressure vessels in 13 Russian designed plants in Russia and Eastern Europe have been successfully annealed. This experience lends significant credibility that thermal annealing in the U.S. is feasible from an engineering point of view.

The engineering feasibility of thermal annealing is being addressed through two Annealing Demonstration Projects, being jointly funded by the U.S. Department of Energy and the nuclear industry, including some international support. Currently, the two demonstration projects are planned using the reactor pressure vessels of two canceled PWR plants. One plant, Marble Hill in Indiana, is a typical Westinghouse design, while the other, Midland in Michigan, is a typical B&W [Babcock and Wilcox] design.

HUMAN FACTORS SESSION

L. F. Hanes and J. F. O'Brien presented a paper about the human factors program at Electric Power Research Institute (EPRI).[6] They indicated that EPRI has produced 67 products since the program began in 1975. They then reviewed the program products, noting that more than 60% of them concern maintenance. This, however, reflects more recent efforts rather than a trend. They also noted

> The HPT [Human Performance Technology] PG [Product Group] thrust has changed over the years with shifting utility needs. Products in pre-TMI years were concerned with control room reviews and methods for correcting human factors deficiencies. Following TMI, products addressing control room human factors issues were expanded. Products were delivered dealing with control room enhancements, displays, and alarm systems. A Human Factors Primer was published to make nuclear plant management more aware of the need for and value of human factors. During the immediate post-TMI period, human factors products addressing maintenance issues were completed, also. The interest in maintenance resulted from recognition that the majority of consequential human errors occur during maintenance tasks, that high costs are associated with such errors, and that worker training is expensive. In recent years, the EPRI HPT program has shifted mainly to products solving maintenance problems.[6]

Another paper[7] presented the results of a design review of an advanced human-system interface evaluation. That paper is part of the draft NUREG-0700, Rev. 1, and notes, "The NRC has established programs to review the human factors engineering (HFE) aspects of design and implementation of significant changes to existing CRs [control rooms] and advanced CR designs in order to help assure that the incorporation of advanced technology enhances the potential safety benefits and minimizes the potentially negative effects on performance and plant safety."[7]

Finally, a paper presented in the PRA topics session concerned Human Reliability Analysis (HRA) methods.[8] That paper focused on one aspect of HRA, "the omission of errors of commission, or those errors that are associated with inappropriate interventions by operators with operating systems."[8] A team of people from several organizations (Battelle National Laboratory, John Wreathall & Company, Science Applications International Corporation, and Buttonwood Consulting, Inc.) sought "to develop a new method for HRA based on analyzing risk-significant operating experience."[8] The project was divided into four phases: (1) Assessment Phase, (2) Analysis and Characterization Phase, (3) Development Phase, and (4) Implementation Phase. The first two phases are complete, and the results were published in NUREG/CRs.[9,10] The third phase is the subject of the presented paper, and

> . . . the concepts of the framework have matured into a working HRA method, with identified process steps. This working HRA method, albeit in preliminary form, has been expanded by using trial applications concluding in quantification of a human failure event. The new HRA method, called ATHEANA (A Technique for Human Error Analysis), improves the ability of PRAs to:
>
> - identify and characterize important human-system interactions and their likely consequences under accident conditions;
> - represent the most important severe accident sequences that could occur;
> - estimate the frequencies of these sequences and the associated probabilities of human errors; and
> - provide recommendations for improving human performance based upon characterizations of the causes of human errors.[8]

The authors found that

> The trial application was a 'proof of concept' for ATHEANA; it demonstrated that it is possible to identify and estimate the probabilities of HFEs [and associated EFCs (error forcing contexts)] that have an observable impact on the frequency of core damage, and which are generally not included in current PRAs.
>
> A general process was outlined that addresses the iterative steps of defining HFEs and estimating their probabilities using search schemes.
>
> A knowledge base was developed with the objective of describing the links between unsafe actions and error forcing contexts, and is based on behavioral science theory and analysis of NPP [nuclear power plant] events.[8]

PROBABILISTIC RISK ASSESSMENT TOPICS

Two papers from researchers at Idaho National Engineering Laboratory (INEL)[11,12] in association with NRC and a third[13] written in conjunction with Sandia National Laboratories (SNL) and NRC characterize the PRA topics theme. All three papers concern the use of computer codes developed by INEL under the auspices of the NRC. These codes are grouped together and included in a suite that is defined in the paper:

> The Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE), refers to a suite of computer programs that were developed to create and analyze a probabilistic risk assessment (PRA) of a nuclear power plant. The programs in this suite include: Models and Results Data Base (MAR-D) software, Integrated Reliability and Risk Analysis System (IRRAS) software, Systems Analysis and Risk Assessment (SARA) software, Fault tree, Event tree, and Piping and instrumentation diagram (FEP) graphical editor, and the Graphical Evaluation Module (GEM) software. Each of these programs performs a specific function in taking a PRA from the conceptual state all the way to publication.[11]

The first paper[11] discusses the features and capabilities of the SAPHIRE computer codes. New software development implementing the Windows-based 32-bit version is also presented. The suite of programs now support Level 2 PRA analyses as well as allow an unlimited number of basic events, gates, systems, and event trees in the data base. Older versions of the data are automatically converted to a newer version; however, because of the increase in the number of events, gates, and systems, the data base has essentially doubled in size. The next two papers regard the use of SAPHIRE for specific purposes. INEL developed Level 1 risk models for use with SAPHIRE for the PRA plant-specific Accident Sequence Precursor (ASP) Program.[12] To better understand the work by INEL, a few basic definitions from the report are presented here:

> - Precursor. In general, an accident sequence precursor is a sequence of events that successfully prevented core damage, that if additional failures had occurred, would have resulted in core damage. Precursors have been separated into two types. The first type involves the occurrence of an initiating event (analyzed in an initiating event assessment), and the second type involves failures or degradations in mitigating equipment (analyzed in a condition assessment). In the ASP Program, a precursor is retained if the likelihood* of those additional failures leading to core damage is greater than or equal to 1.0×10^{-6}. Occasionally, there are exceptions for events with coincident containment failure or unique influences difficult to analyze (sic). These are reported in the ASP Program annual reports but do not necessarily receive the same level of analytical treatment as the more traditional precursors.
>
>   *Likelihood is used in this context to indicate either a conditional core damage probability (CCDP) or a change in CCDP (i.e., event importance).
> - Change Set. A change set is a listing of the risk model basic events that have been designated with a change in probability from the base case. In ASP evaluations, a change set is used to identify the basic events (and initiating events) that must be re-evaluated to represent the events and conditions observed during the reported precursor. GEM assists in creating the proper ASP evaluation change set by automatically identifying basic events that must change once the initiating event or condition duration is specified.
> - Conditional Core Damage Frequency (CCDF). The conditional core damage frequency is the hazard rate

representing the expected number of core damage events per hour given a set of known failures or plant operating conditions.

- Conditional Core Damage Probability (CCDP). The conditional core damage probability is the likelihood of experiencing a core damage event given a set of known failures or plant operating conditions. When calculated for an operational event, the CCDP is a measure of how close the plant came to core damage during the event. Alternatively, the CCDP can be thought of as the likelihood of failure of the remaining barriers to core damage.

INEL's work focused on four areas: (1) model improvements, (2) revision 1 models, (3) revision 2 models, and (4) ASP model extension. The first and last areas profile the scope of work, whereas the middle two areas detail specific changes and are also the subject of other reports. Excerpts from the presented paper concerning areas 1 and 4 are included here.

MODEL IMPROVEMENTS

INEL investigated the feasibility of making enhancements in several modeling areas. These areas were:

- Uncertainty Analysis. The ASP models have never had the ability to give an uncertainty estimation. It was well-known that a basic parameter uncertainty estimation capability comparable to that of a typical full-scope PRA was necessary and practical. The INEL was also tasked with investigating how to estimate the unique modeling uncertainty associated with simplified ASP models.
- Human Reliability Analysis. The purpose of this task was to make improvements in the current practice for human reliability analysis (HRA) for the ASP program. Specific areas needing attention were the treatment of recovery errors and the assessment of dependency. The goal was to develop a general, easy-to-apply, method which handled actuation, recovery, and dependency through a consistent model of human behavior.
- Common Cause Failure [CCF] Analysis. The CCF improvements work focused on providing better basic parameter estimates while not increasing the complexity of the models. The current ASP logic models are straightforward for construction and review purposes and they generate a reasonable number of simple cutsets . . . .

  Since the ASP logic models were to remain unchanged, the focus was placed on the CCF basic events values. Use of the Multiple Greek Letter method was simple for point estimate calculations but was complicated for uncertainty analysis. Therefore, other alternatives were investigated, with conversion to the Alpha method being the final determination . . . .

- Modeling Level of Detail and Scope. . . . Enhancements were prioritized and it was decided to develop support system models as part of this work with external events, low power/shutdown [LP/SD] risk, and Level 2 and 3 risk models . . . .

  Each improvement was demonstrated on several plant models selected from a set of prototype models consisting of Byron, St. Lucie, Peach Bottom, Oconee, and Three Mile Island . . . .

ASP MODEL EXTENSION

. . . RES contracted the INEL to perform a feasibility study to investigate extending the ASP models, giving them the capability of analyzing the risk significance of operational events associated with LP/SD operations, full-power and LP/SD internal flooding, LP/SD fire and earthquakes. Full-power fire and earthquakes were already being evaluated by another project . . . .

Specific selection guidance is being developed in the following areas:

(1) Low Power Operations. The selection guidance used for full power operation is conservatively being used for low power operations.

(2) Shutdown Operations. The selection guidance addresses various situations involving loss of shutdown cooling, loss of coolant inventory control, and conditions that could impede proper operator actions.

(3) Full Power External Events. The selection guidance addresses actual occurrences of an external event and various situations where conditions are such that given an external event, the CCDP would likely be greater than 1.0×10^{-6}.

(4) Shutdown External Events. The selection guidance for these events is an adaptation of that for shutdown internal events, with the external events viewed as another means of impacting shutdown cooling, inventory control, and operator actions.

INEL, in conjunction with SNL under the auspices of NRC,[13] used previous work[11] to develop Level 2 risk models for use in the ASP Program. Their objective is outlined as follows:

As outlined in the ASP Program Plan, the ASP Program pursues the ultimate objective of performing risk significant evaluations on operations events (precursors) occurring in commercial nuclear power plants (NPPs). To achieve this objective, the Office of Nuclear Regulatory Research (RES) is supporting the development of simple probabilistic risk assessment (PRA) models for NPPs in the U.S. Presently, only simple Level 1 plant models have been developed which estimate core damage frequencies. However, the plan calls for the capability to append to existing Level 1 outcomes the capability of performing Level 2/3 risk assessments such that the potential consequences and risks could also be assessed.

The paper goes on to explain the scope of work necessary to accomplish their objectives:

> This paper documents the current ASP Level 2/3 model development effort. As done in the Level 1 model development, all NPPs are classified into groups with a single plant being selected from each group as the subject of the initial model development effort. During the plant-group model development, to the extent feasible, information and methods developed and collected in the course of the NUREG-1150 study [NRC, 1990] are utilized. The objectives of the Level 2/3 plant-group model development are to demonstrate: (1) appropriate interfaces between the Level 1 models and the Level 2 models; (2) simplified Level 2 models; (3) source term (ST) estimates; (4) consequence estimates; and (5) integration of the Level 2/3 models into the existing ASP software.

The paper explains further, "To estimate consequences and risk, a linked event tree approach is utilized where event trees are used to model the Level 1, Level 2, and Level 3 portions of the analysis. Whereas the endstates from the Level 1 tree represent the frequency of core damage for an operation event, the endstates from the Level 3 tree represent its risk." Moreover, "[e]mbedded in this methodology is the flexibility to link Level 2/3 models of various levels of detail to the Level 1 models."

ECCS STRAINER BLOCKAGE RESEARCH AND REGULATORY ISSUES

A paper presented by NRC[14] staff profiles the issues, history, and efforts to date regarding the potential for ECCS strainer blockage. The report explains that historically,

> Nuclear power reactor licensees in the United States of America (US) are required to ensure long-term cooling to the reactor core in accordance with the Code of Federal Regulations (CFR), specifically 10 CFR 50.46, 'Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors.' Following a postulated loss-of-coolant-accident (LOCA), LOCA generated debris (e.g., thermal insulation), operational debris (e.g., corrosion products), and foreign materials may migrate to the ECCS suction strainer in boiling water reactors (BWRs) and block (clog) the suction strainer. If the resistance across the strainer is great enough, the performance of the ECCS pumps may be degraded because of a decrease in net positive suction head (NPSH) and result in a reduction or elimination of flow available for core cooling and decay heat removal. Debris blockage of ECCS suction strainers is a safety concern.
>
> The United States Nuclear Regulatory Commission (NRC) first addressed this concern as part of the resolution of Unresolved Safety Issue (USI) A–43, 'Containment Emergency Sump Performance,' in the mid 1980s.

However, more recently,

> Events at two operating reactors in 1992 and 1993 compelled the NRC to re-evaluate the debris blockage aspects of USI A–43 for BWRs and its resolution of the issue.

The paper continues,

> On July 28, 1992, a safety relief valve (SRV) inadvertently opened in the Barsebäck-2 nuclear power plant, a Swedish BWR, and discharged into the drywell. The steam jet stripped fibrous insulation from adjacent pipes and part of that insulation debris was transported to the wetwell pool and accumulated on two strainers. After about one hour, the resultant increase in head loss from debris blockage across the strainers caused the plant operators to stop pump operations to prevent damage to the pumps. The plant operators backflushed both strainers, then reinitiated containment spray.

The paper further indicates that two additional events occurred:

> On January 16, and April 14, 1993, two events involving the clogging of ECCS strainers occurred at the Perry nuclear power plant, a US BWR. The first Perry event involved clogging of residual heat removal (RHR) pump suction strainers by debris in the suppression pool. The second Perry event involved deposition of fibers from temporary drywell cooling filter materials inadvertently (i.e., foreign material) dropped into the pool on these strainers coupled with the filtration of suppression pool particulates (i.e., corrosion products or 'sludge').

The insights from the analyses of the Barsebäck and Perry incidents were examined, and the NRC concluded,

> The lessons of Barsebäck and Perry have demonstrated that the ECCS is susceptible to a potential common-cause failure due to clogging of the suction strainers that could prevent them from being able to perform their safety function of decay heat removal over the long term. This lesson was reinforced by a recent strainer clogging event that occurred at Limerick, Unit 1 on September 11, 1995.
>
> A stuck open safety relief valve at Limerick led to the initiation of both loops of suppression pool cooling. The 'A' loop subsequently experienced flow and current oscillations, an indication of a clogged strainer. Limerick operators shutdown and restarted the pump with no further complications, and the rest of the event was mitigated without any additional problems. Post-event strainer inspection revealed that the suction strainer had been clogged by a combination of polyethylene fibers and corrosion products (sludge). The sludge is generally present in US BWRs; however, the fibrous material that was present in the suppression pool was due to a failure

at some time to prevent the introduction of foreign material into the suppression pool.

As a result, the NRC reevaluated its position on USI A–43:

> Because of the 1992 Barsebäck event, the 1993 Perry events, and the actions and research of Swedish and Finnish regulatory authorities, the NRC began to evaluate the effects of debris blockage of suction strainers on the long-term cooling function of the ECCS system. The purpose of the evaluation was to determine if strainer blockage was a safety concern for US BWRs and to judge the adequacy of the resolution of USI A–43 as it pertains to BWRs. The initial thrust of this study was both probabilistic and deterministic, with emphasis on estimating the probability of losing NPSH margin.

Regarding the new evaluation, NRC determined that,

> . . . the staff's analysis and research indicated that debris blockage for BWRs was a greater safety concern than early analysis had concluded.

The paper then reports on the present status of the issue:

> The current state of knowledge regarding the US BWR ECCS strainer blockage issue leads to the following observations and conclusions:
>
> (1) A singular (or generic) solution is not possible in the US because of the variations in BWR containment designs, installed passive strainers, ECCS long term cooling requirements, variability of installed insulation materials, and foreign materials present . . . .
>
> (2) Calculational models to accurately predict amounts of debris generated and the physical characteristics of such debris are lacking . . . .
>
> (3) Drywell transport of LOCA generated debris is the second area of continuing technical debate . . . .
>
> (4) Suppression pool debris transport models which include settling and materials dependence (to a varying degree) have been developed. There appears to be consensus that suppression pool debris transport modeling, as described in Reference 8, is acceptable and reasonably well understood . . . .
>
> (5) Experiments over the past 3–4 years have produced considerable data to estimate pressure drop associated with debris materials which can be transported to the suction strainers . . . .

The paper also points out how the agency is addressing outstanding issues:

> In retrospect, it is clear that prior USI A–43 evaluations and findings developed in the mid 1980s, when compared to the current knowledge base and recent plant incidents, will under-estimate the potential for loss of long term cooling capability during the post-LOCA period for US BWRs. Based on the current knowledge base and available plant information the conclusion was reached that passive suction strainers currently installed in US BWRs are probably undersized and susceptible to the detrimental effects of debris blockage. The NRC staff considers larger strainer surface areas the best solution to the BWR Suction Strainer Blockage Issue.

The paper specified the issues: "There are two aspects to the potential strainer clogging issue. The first is the potential for clogging of ECCS suction strainers by debris and sludge which is present in the pool during normal operation." Consequently, the paper points out, "Based on these considerations, NRC Bulletin 95–02 was issued . . . ." The other issue addressed by the paper is explained: "The second aspect of the issue is the potential clogging of ECCS suction strainers by a combination of debris generated by a LOCA and debris and sludge which is present in the pool during normal operation." Following this, the paper indicates, "The staff issued a draft bulletin and proposed revision to Regulatory Guide 1.82, Rev. 1 for public comment . . . ."

Another paper presented at the conference[15] dealt with the international repercussions resulting from the Barsebäck incident:

> The Barsebäck incident spurred immediate action on the part of regulators and utilities alike in several Organization for Economic Cooperation and Development (OECD) countries (e.g., Sweden, Finland, Germany, Switzerland, and France). For example, the Swedish Nuclear Power Inspectorate (SKI) required that immediate measures to prevent strainer clogging should be taken for the five oldest Swedish BWRs which had strainers of small area before they were allowed to start again.

The paper also indicated that several steps were initially taken to improve communications and dissemination of information,

> To accelerate exchange of information and experience, and provide feedback of actions taken to the international community, a workshop on the strainer clogging issue was hosted by SKI in Stockholm, Sweden, on January 26–27, 1994, under the auspices of Committee on the Safety of Nuclear Installations/Principle Working Group–1 (CSNI/PWG–1). The objectives of the workshop were (1) to give an overview of decisions and work performed recently on this issue, (2) to address the actual safety issues with regard to the reliability of ECC recirculation, and (3) to discuss further actions needed. The workshop revealed a rather confusing picture of the available knowledge base, examples of conflicting information and a wide range of interpretation of guidance provided in the USNRC Regulatory Guide 1.82, Rev. 1. Following this workshop, SKI requested formation of an international working group (IWG) under the auspices of

CSNI/PWG–1 committee for establishing an internationally agreed-upon knowledge base for assessing the reliability of emergency core cooling water recirculation systems.

The specific tasks given to the group were:

(1) Critical review and compilation of available experiments and other data related to the performance of ECC water recirculation systems, including formation and behavior of various types of debris contaminating the water.

(2) Assessment of the applicability of the data base. Identification of major uncertainties, lack of information and data.

(3) Proposal of additional research and experiments as well as pointing out those uncertainties which should be accommodated in terms of conservative design features.

The IWG, composed of participants from German (GRS), Swedish (SKI), Finnish (STUK), Japanese (NUPEC), and US (USNRC) regulatory authorities, the US BWR Owners Group (BWROG), insulation vendors (PCI and Transco Products, Inc.), Vattenfall Utveckling AB, and SEA [Science and Engineering Associates, Inc.] (a NRC subcontractor), met initially in April 1994 in Stockholm, Sweden, and three additional meetings have been held.

The IWG indicated that "[t]wo design approaches have emerged for dealing with LOCA generated debris—the robust design approach (the Nordic countries) and the 'calculational' approach (United States)." The IWG divided the LOCA considerations to be addressed in ECCS design into five areas: Debris Generation, Drywell Transport, Wetwell Transport, Strainer Pressure Drop, and Related Issues. Their conclusions regarding these areas are as follows:

Debris Generation:	". . . the IWG concludes that plant-specific studies are needed."
Drywell Transport:	". . . conservative assumptions are recommended regarding the fraction of the debris transported through the drywell."
Wetwell Transport:	"The new experiments using debris that was removed from the reactors or aged by temperature, showed that the material tended to remain suspended in the water and thus is available for strainer clogging."
	"Small particles, in combination with fibrous debris, would generally promote strainer clogging."
Strainer Pressure Drop:	"Many experiments were performed on mixtures of fibers and particles. Generally, the pressure drops significantly increased for these mixtures as compared to pure materials."
	"Experiments also showed that the regression curves used for fibrous insulation in the earlier guidelines were non-conservative."
	"A mixture of fibrous and reflective metallic insulation [RMI] debris have been found to result in head losses that are higher than the sum of their pure constituents."
Related Issues:	"A number of potential concerns for the recirculation operability were identified that encompass such issues as vent path clogging; generation of missiles; strainer penetration; potential for liquid flow restrictions in, for instance, spray nozzles and fuel bundles; and effects on pump operability."

A third paper presented at the conference[16] reports on the computer code that was developed in support of the NRC's studies of the ECCS strainer blockage phenomenon. SEA, under the auspices of the NRC, was selected to "estimate the potential for loss of net positive suction head (NPSH) margin of the ECCS pumps in a BWR due to clogging of suction strainers by a combination of fibrous and particulate debris, either generated by the LOCA or previously present inside the containment." The results of SEA's work were published in a NUREG/CR.[17]

SEA determined that

NUREG/CR–6224 concluded that for the reference plant, a BWR 4 with a Mark 1 containment whose dry well piping is essentially all insulated with steel jacketed fibrous material, considerable potential exists for loss of the ECCS pumps due to strainer clogging by the LOCA generated debris. It was further concluded that very thin insulation debris layers are sufficient to induce pressure drops that are in excess of available NPSH margin for most BWRs.

Further, SEA reported,

The NUREG/CR–6224 methodology was codified in a computer code named BLOCKAGE. The elements of the methodology developed for BLOCKAGE to evaluate the effect on head loss across strainers due to debris introduced into the suppression pool as a result of a LOCA follows the key LOCA event progression phenomenology . . . .

Moreover,

A LOCA event can be effectively divided in two phases for the purpose of BLOCKAGE analysis: (1) the short term phase, starting with the actual weld failure and ending with the drywell de-pressurization and (2) the long term phase which ends when the ECCS and long term residual heat removal functions are no longer necessary. BLOCKAGE can be effectively set-up to analyze both phases of a LOCA event.

CONCLUSIONS

It is not possible to summarize or even mention all the papers presented at the conference. Nor is it possible to provide an adequate review of the meeting; therefore the reader is urged to read all the papers published in the proceedings and again is referred to Table 1, which lists all the reports.

Table 1 List of Proceedings of the Twenty-Third Water Reactor Safety Information Meeting[a,b]

	PLENARY SESSION AND LUNCHEON SPEECH	
1.	The Role of Research in NRC Regulatory Programs	S. A. Jackson (NRC)
2.	Current Issues with Research Support	W. T. Russell (NRC)
3.	Annealing the Reactor Vessel at the Palisades Plant	R. A. Fenech (Consumers Power Company)
4.	Need for Higher Fuel Burnup at the Hatch Plant	J. T. Beckham (Southern Nuclear Operating Company)
5.	Luncheon Remarks	J. M. Taylor (NRC)
	HIGH BURN-UP FUEL BEHAVIOR	
	Chairperson: R. Meyer	
1.	New Results from Pulse Tests in the CABRI Reactor	F. Schmitz, J. Papin, and M. Haessler (CEA/IPSN); N. Waeckel (EdF)
2.	New Results from the NSRR Experiments Reactor with High Burnup Fuel	T. Fuketa et al. (JAERI)
3.	Recent View to the Results of Pulse Tests in the IGR Reactor with High Burn-up Fuel	V. Asmolov and L. Yegorova (RRC/KI)
4.	High Burnup Effects in WWER Fuel Rods	V. Smirnov and A. Smirnov (RRC/RIAR)
5.	Assessment of Reactivity Transient Experiments with High Burnup Fuel	O. Ozer and R. Yang (EPRI); Y. Rashid and R. Montgomery (ANATECH)
6.	Power Excursion Analysis for BWRs at High Burnup	D. Diamond, L. Neymotin, and P. Kohut (BNL)
7.	Review of Halden Reactor Project High Burn up Fuel Data That Can Be Used in Safety Analyses	W. Wiesenack (OECD Halden Project)
8.	New High Burnup Fuel Models for NRC's Licensing Audit Code, FRAPCON	D. Lanning, C. Beyer, and C. Painter (PNL)
	THERMAL HYDRAULIC RESEARCH	
	Chairperson: W. Hodges	
1.	GIRAFFE Test Results Summary	S. Yokobori, K. Arai, and H. Oikawa (Toshiba)
2.	Results from the NRC AP600 Testing Program at the Oregon State University APEX Facility	J. Reyes (OSU), D. Bessette (NRC), and M. DiMarzo (UM)
3.	PUMA Test Program for SBWR	M. Ishii et al. (PU)
4.	The PANDA Tests for SBWR Certification	G. Varadi et al. (PSI)
5.	NRC Confirmatory AP600 Safety System Phase I Testing in the ROSA/AP600 Test Facility	G. Rhee (NRC), Y. Kukita (JAERI), and R. Schultz (INEL)

(Table continues on the next page.)

Table 1 (Continued)

HUMAN FACTORS RESEARCH

Chairperson: J. Persensky

1.	Review of EPRI Nuclear Human Factors Program	L. Hanes and J. O'Brien (EPRI)
2.	Interim Results of the Study of Control Room Crew Staffing for Advanced Passive Reactor Plants	B. Hallbert, A. Sebok, and K. Haugset (OECD Halden Project); D. Morisseau and J. Persensky (NRC)
3.	Human-System Interface Design Review Guideline: The Development of Draft Revision 1 to NUREG-0700	J. O'Hara, W. Stubler, and W. Brown (BNL); J. Wachtel and J. Persensky (NRC)

ADVANCED I&C HARDWARE AND SOFTWARE

Chairperson: C. Antonescu

1.	Lessons Learned from Development and Quality Assurance of Software Systems at the Halden Project	T. Bjorlo et al. (OECD Halden Project)
2.	Assessment of Fiber Optic Sensors and Other Advanced Sensing Technologies for Nuclear Power Plants	H. Hashemian (AMS)
3.	Preliminary Studies on the Impact of Smoke on Digital Equipment	T. Tanaka (SNL), K. Korsah (ORNL), and C. Atonescu (NRC)
4.	Development of Electromagnetic Operating Envelopes for Nuclear Power Plants	P. Ewing and S. Kercel (ORNL)
5.	Performance Evaluation of Fiber Optic Components in Nuclear Plant Environments	M. Hastings and D. Miller (Ohio State U.); R. James (EPRI)

SEVERE ACCIDENT RESEARCH I

Chairperson: C. Tinkler

1.	Resolution of the Direct Containment Heating Issue for All Westinghouse Plants with Large Dry Containments or Subatmospheric Containments	M. Pilch, M. Allen, and E. Klamerus (SNL)
2.	Status of the FARO/KROTOS Melt-Coolant Interactions Tests	D. Magallon et al. (JRC, Italy)
3.	An Overview of Fuel-Coolant Interactions (FCI) Research at NRC	S. Basu and T. Speis (NRC)
4.	Progress on the MELCOR Code	K. Bergeron et al. (SNL)
5.	Investigation of a Steam Generator Tube Rupture Sequence Using VICTORIA	N. Bixler and C. Erickson (SNL); J. Schaperow (NRC)
6.	The Severe Accident Research Programme PHEBUS F. P.: First Results and Future Tests	M. Schwarz (IPSN/CEA) and P. von der Hardt (JRC, France)

SEVERE ACCIDENT RESEARCH II

Chairperson: A. Rubin

1.	Preliminary Results of the XR2-1 Experiment	R. Gauntt and P. Helmick (SNL); L. Humphries (SAIC)
2.	Steady-State Observations and Theoretical Modeling of Critical Heat Flux Phenomena on a Downward Facing Hemispherical Surface	F. Cheung and K. Haddad (PSU)
3.	Hydrogen Detonation and Detonation Transition Data from the High-Temperature Combustion Facility	G. Ciccarelli et al. (BNL), H. Tagawa (NUPEC), and A. Malliakos (NRC)
4.	Recent Experimental and Analytical Results on Hydrogen Combustion at RRC "Kurchatov Institute"	S. Dorofeev et al. (RRC/KI)
5.	SCDAP/RELAP5 Code Development and Assessment	C. Allison and J. Hohorst (INEL)
6.	Recent SCDAP/RELAP5 Improvements for BWR Severe Accident Simulations	F. Griffin (ORNL)

Table 1 (Continued)

	PROBABILISTIC RISK ASSESSMENT TOPICS	
	Chairperson: M. Cunningham	
1.	Development of an Improved HRA Method: A Technique for Human Error Analysis (ATHEANA)	J. Taylor and W. Luckas (BNL), J. Wreathall (JWC), S. Cooper (SAIC), and D. Bley (BCI)
2.	Uncertainties in Offsite Consequence Analysis	M. Young and F. Harper (SNL); C. Lui (NRC)
3.	Advanced Accident Sequence Precursor Analysis Level 1 Models	M. Sattison et al. (INEL)
4.	Advanced Accident Sequence Precursor Analysis Level 2 Models	W. Galyean, D. Brownson, J. Rempe (INEL); T. Brown, J. Gregory, F. Harper (SNL); C. Lui (NRC)
5.	New Developments in the SAPHIRE Computer Codes	K. Russell, S. Wood, and K. Kvarfordt (INEL)
	INDIVIDUAL PLANT EXAMINATION	
	Chairperson: T. Su	
1.	Core Damage Frequency Perspectives for BWR 3/4 and Westinghouse 4-Loop Plants Based on IPE Results	S. Dingman and A. Camp (SNL), J. LaChance (SAIC), and M. Drouin (NRC)
2.	Severe Accident Progression Perspectives for Mark I Containments Based on the IPE Results	C. Lin, J. Lehner, and W. Pratt (BNL); M. Drouin (NRC)
3.	Perspectives on Plant Vulnerabilities and Other Plant and Containment Improvements	J. LaChance, A. Kolaczkowski, and J. Kahn (SAIC); R. Clark and J. Lane (NRC)
4.	IPE Results as Compared with NUREG-1150	W. Pratt and J. Lehner (BNL), A. Camp (SNL), and E. Chow (NRC)
5.	IPE Data Base: Plant Design, Core Damage Frequency and Containment Performance Information	J. Lehner, C. Lin, and W. Pratt (BNL); T. Su and L. Danziger (NRC)
	STRUCTURAL & SEISMIC ENGINEERING	
	Chairperson: J. Costello	
1.	An Assessment of Seismic Margins in Nuclear Plant Piping	W. Chen and K. Jaquay (ETEC); N. Chokshi and D. Terao (NRC)
	PRIMARY SYSTEMS INTEGRITY	
	Chairperson: M. Mayfield	
1.	RPV and Steam Generator Pressure Boundary	J. Strosnider (NRC)
2.	Environmentally Assisted Cracking of LWR Materials	O. Chopra et al. (ANL)
3.	Steam Generator Tube Integrity Program	D. Diercks and W. Shack (ANL); J. Muscara (NRC)
4.	Embrittlement Recovery Due to Annealing of Reactor Pressure Vessel Steels	E. Eason, J. Wright, and E. Nelson (MCS); G. Odette and E. Mader (UCSB)
5.	Reactor Pressure Vessel Integrity Research at the Oak Ridge National Laboratory	W. Corwin, W. Pennell, and J. Pace (ORNL)
	EQUIPMENT OPERABILITY AND AGING	
	Chairperson: J. Vora	
1.	Condition Monitoring and Testing for Operability of Check Valves and Pumps	D. Casada and K. McElhaney (ORNL)
2.	Corrosion Effects on Friction Factors	L. Magleby (INEL) and S. Shaffer (Battelle)
3.	Results of a Literature Review on the Environmental Qualification of Low-Voltage Electric Cables	R. Lofaro, B. Lee, and M. Villaran (BNL); J. Gleason (GLS); S. Aggarwal (NRC)
4.	DOE-Sponsored Cable Aging Research at Sandia National Laboratories	K. Gillen et al. (SNL)
5.	DOE-Sponsored Aging Management Guideline for Electrical Cable and Terminations	G. Gazdzinski (OEES)

(Table continues on the next page.)

Table 1 (Continued)

ECCS STRAINER BLOCKAGE RESEARCH AND REGULATORY ISSUES

Chairman: C. Serpan

1.	An Overview of the BWR ECCS Strainer Blockage Issues	A. Serkiz, M. Marshall, and R. Elliott (NRC)
2.	The CSNI/PWG-1 International Task Group on ECCS Reliability	O. Sandervåg (SKI, Sweden), T. Riekert (GRS, Germany), A. Serkiz (NRC), and J. Hyvärinen (STUK, Finland)
3.	Experiments of ECCS Strainer Blockage and Debris Settling in Suppression Pools	G. Hecker et al. (ARL)
4.	The Stainer Blockage Assessment Methodology Used in the BLOCKAGE Code	G. Zigler and D. Rao (SEA)

[a]Abbreviations of organizations are as follows:

AMS	Analysis and Measurements Services Corp.
ANL	Argonne National Laboratory
ARL	Alden Research Laboratory
BCI	Buttonwood Consulting Inc.
BNL	Brookhaven National Laboratory
EPRI	Electric Power Research Institute
ETEC	Energy Technology Engineering Center
GLS	GLS Enterprises, Inc.
INEL	Idaho National Engineering Laboratory
IPSN/CEA	French Atomic Energy Commission
JAERI	Japanese Atomic Energy Research Institute
JRC	European Commission, Joint Research Centre
JWC	John Wreathall & Co.
MCS	Modeling and Computing Services
NRC	Nuclear Regulatory Commission
NUPEC	Nuclear Power Engineering Corp.
OECD	Organization for Economic Cooperation and Development
OEES	Ogden Environmental Energy Services
ORNL	Oak Ridge National Laboratory
OSU	Oregon State University
PNL	Pacific Northwest Laboratory
PSI	Paul Scherrer Institute
PSU	Pennsylvania State University
PU	Purdue University
RRC/KI	Russian Research Centre–Kurchatov Institute
RRC/RIAR	Russian Research Centre–Research Institute of Atomic Reactors
SAIC	Science Applications International Corp.
SEA	Science and Engineering Associates, Inc.
SNL	Sandia National Laboratories
UCSB	University of California, Santa Barbara
UM	University of Maryland

[b]All papers and talks are included in the proceedings.

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Accident Analysis

Edited by R. P. Taleyarkhan

Analysis of a PWR LBLOCA Without SCRAM[a]

By Trevor N. Tyler, Rafael Macian-Juan, and John H. Mahaffy[b]

Abstract: *We analyze a conservative recriticality scenario to explore the potential risk of fuel damage during a large-break loss-of-coolant accident in a typical U.S. pressurized-water reactor. No SCRAM is assumed, and no credit is taken for injected boron in core neutronics calculations. Although the scenario is conservative, the analysis is best estimate, using TRAC-PF1/MOD2 to model the thermal–hydraulics, coupled with a three-dimensional, transient neutronic model of the core. The simulation can follow complex system interactions during the reflood, which influence the neutronic feedback in the core. In all cases examined, the return of cold water to the core is limited by increased steam production from a marginal (local) return to power. A quasi-steady state is established during low-pressure safety injection cooling in which sufficient core flow exists to maintain rod temperatures to well below the fuel damage limit, but insufficient total inventory is present to result in a full return to power.*

The risk of reactivity accidents has been considered an important safety issue since the beginning of the nuclear power industry. In particular, several events leading to such scenarios for pressurized-water reactors (PWRs) have been recognized and studied to assess the potential risk of fuel damage. The common characteristic of such events is the injection of cold water with low boron content into the core. Depending on the thermal–hydraulic conditions during the injection, the potential exists for a return to criticality, which could lead to a dangerous power excursion.

Some studies have been performed to analyze this type of accident during local dilution transients[1] and small-break loss-of-coolant accidents (SBLOCAs),[2] and the potential for a recriticality scenario following the large-break loss-of-coolant accident (LBLOCA) reflood phase has also been identified. One of the common conclusions reached in these studies has been the necessity of a detailed three-dimensional (3-D) thermal–hydraulic and a 3-D neutronic core model to better describe the transient evolution. A version of TRAC-PF1/MOD2 has been created with these features at The Pennsylvania State University.

The work described here analyzes one of these accidents, the LBLOCA, from its initiation to the postreflood cooling phase with the use of TRAC-PF1/MOD2 3-D core and vessel models. This study is an extension of LBLOCA analyses made by Los Alamos National Laboratory[3,4] with the use of a modified version of TRAC-PF1/MOD2v5.3. In these studies, a reactor SCRAM was assumed, and the potential for recriticality was thus averted. For analysis described in this article, a similar plant model was used, but two major differences were introduced: no SCRAM was simulated, and a full 3-D neutronic core model was used instead of the classic point-kinetics model. As a conservative assumption, no credit was taken for boron in the emergency core cooling system (ECCS). The fine thermal–hydraulic and neutronic coupling allowed

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a refined study of the core power evolution as the transient progressed and could predict a return to a critical state should it occur during the reflooding phase.

This work also differs from the Los Alamos study because a modified version 5.4 of TRAC-PF1/MOD2 was used rather than version 5.3. The modifications to the thermal–hydraulic portion of the code are essentially identical to those used by Los Alamos in the analysis with version 5.3. They are required to correct problems with water packing and interfacial drag logic in the downcomer that prevent either version 5.3 or 5.4 from successfully running this class of LBLOCA transient.

Our purpose in this study is not to produce a high fidelity reproduction of conditions in a specific plant but to explore the general consequences of feedback between core neutronic and thermal–hydraulic behavior. We will demonstrate that feedback between moderator-induced power increases, and the resulting increase in core pressure is rapid enough to keep the core liquid level below that required for a full return to power.

This article first presents the generic system model together with the assumptions involved in its development, especially in the neutronics part. Then the computational process is described, and the results of the steady state are presented. Finally, a description of the transient is followed by the conclusions.

COMPUTER MODEL DESCRIPTION

Thermal–Hydraulic Model

The thermal–hydraulic plant model used in this study was developed as part of the international safety research effort known as the 2-D/3-D Refill/Reflood Program. The model describes a typical U.S./Japan PWR with four loops and 949 fluid cells.

The primary side is modeled in detail except for the high- and low-pressure safety injection (HPSI and LPSI) systems that are simulated by four mass flow boundary conditions (TRAC Fill components) representing the combined HPSI and LPSI in each loop. Four active accumulators (one per loop), modeled by means of pipe and valve components, complete the ECCS. The nitrogen in the accumulators is allowed into the system and followed by TRAC as an additional gas field. Therefore its effect on condensation rates within the primary-side components is taken into account.

The four loops of the primary side are identical except for the cold leg of the loop 3, where the 200% guillotine break occurs between the cold-leg nozzle and the ECCS injection point. The containment backpressure is simulated by time-dependent boundary conditions implemented with a table option in the TRAC Break component. From an initial atmospheric pressure (0.0999 MPa), the containment pressure reaches a peak of 0.341 MPa at 25 s and then drops to 0.242 MPa at 300 s.

The pressurizer is connected to a hot leg of the second loop. For this work the water level is set to 2.7 m, according to the minimum requirement of Technical Specifications in a PWR. This level was selected as a conservative assumption.

The steam generators (SGs) are modeled with a moderate level of detail. They contain a recirculation path in the secondary side with the recirculation flow rate adjusted at steady state to match typical plant value. The balance of the plant is simulated by mass flow boundary conditions for feedwater flows (TRAC Fills) and pressure boundary conditions at the steam outlets. The secondary-side pressure is set during a steady-state calculation through active control of a valve between the secondary-side outlet nozzle and the Break component, and its value is adjusted to keep the plant energy balance within actual values.

The vessel component is divided into 544 hydrodynamic cells representing 17 axial levels, 4 radial rings, and 8 azimuthal sectors. Most of the vessel internals are included in the model: downcomer, upper and lower plena, upper head, and the spray nozzles are located there. Leakage paths between the hot legs and core barrel are also included.

Finally, the core region within the vessel is contained in the inner two rings and between axial levels five to nine. This region is subdivided such that each inner ring azimuthal sector contains 10.125 fuel assemblies (2065.5 fuel rods), and each outer ring sector contains 14 assemblies (2856 fuel rods). The lower-core support plate covers the three inner radial rings at the top of level four. The upper-core support plate is located at the top of level ten.

References 3 and 4 contain detailed noding diagrams for the TRAC-PF1/MOD2 portion of this plant model.

Neutronic Core Model

The core contains 193 of the 15 by 15 fuel assemblies (a total of 39 372 fuel rods) of three different enrichments: 65 of 2.2 wt %, 64 of 2.7 wt %, and 64 of 3.2 wt %. The mapping of the fuel assemblies and the core hydraulic cells was described in the previous section. The neutronic core model is completed by 64 radial reflector nodes per axial level surrounding the fuel region and 432 reflector nodes in the first and last axial levels to simulate the axial reflectors.

A two-group 3-D neutron kinetics model was developed to simulate the core power evolution during the transient. Previous analyses used a point kinetics model or at best a one-dimensional (1-D) model. The TRAC-PF1/MOD2 version used in this study contains a full 3-D neutronics module developed by Bandini,[5] which is based on the Nodal Expansion Method (NEM)[6] and is capable of handling steady-state and transient situations with several energy groups. The NEM method is one of a variety of recently developed nodal coupling schemes, where the solutions to the transverse integrated 3-D diffusion equation are approximated by a polynomial expansion. This method has been shown to provide accurate results for a variety of benchmark problems.[7,8]

The neutronic core is divided into 368 radial nodes per axial level; each one of the nodes represents the portion of fuel assembly within the axial level except for the center rows of assemblies in the X and Y directions, which are subdivided into four nodes per assembly to ensure symmetry between the thermal–hydraulic and the neutronic noding schemes. This division resulted in a center assembly divided into 16 nodes and in a fine nodalization that allowed fractions of center row fuel assemblies to be mapped with appropriate symmetry into the r-theta distribution of TRAC fuel rod components. The final result was a symmetrical steady-state power mapping consistent with the actual core power distribution of a typical PWR. Figure 1 shows the thermal–hydraulic–neutronic mapping scheme.

The coupling between the neutronic and the thermal–hydraulic core is achieved by 16 active ROD components (one per each core sector in an axial level). Each ROD models an average rod for its respective region and receives the power generated by the sum of all neutron kinetic nodes mapped to it. The axial power shape is determined by the 3-D neutronics calculation. This power is transferred to the core hydraulic cells connected to the active ROD through a standard TRAC heat transfer calculation.[9] The coupling is completed by the information that the rod temperatures and the thermal–hydraulic conditions in the hydraulic core cells feed to the 3-D neutronics module, which is used to obtain the appropriate cross sections for each of the 368 neutronic nodes. This linkage between the neutronics and thermal–hydraulics is numerically explicit and updated at every time step in the calculation.

CALCULATION PROCEDURE

Cross-Section Calculation

Bandini's original NEM implementation in TRAC-PF1/MOD2 used polynomial fitting to obtain the cross sections and diffusion coefficients. This approach works well for transients with relatively small deviations from the normal operating conditions; however, the conditions during an LBLOCA vary widely enough (particularly the mean fluid density) that a more reliable approach is required. For this reason, an interpolation method based on cross-section tables was devised and implemented in the module. Table bounds were established from a baseline LBLOCA with SCRAM. Conditions in the transients reported here always resulted in interpolations within the bounds of our tables.

Only two independent variables were considered to model the two most important reactivity feedback mechanisms expected to control this scenario: the fuel temperature, responsible for the Doppler feedback, and the moderator density, including water temperature and void fraction effects. Coolant density determines the moderation efficiency, which controls the neutron spectrum surrounding the fuel elements. Appropriate consideration of this last effect is very important in this study because the return of liquid to the core is the process expected to lead to recriticality. The 3-D kinetics was selected because nodes located in different parts of the core, even at the same axial level, can be surrounded by different void fractions, which yields a different neutronic behavior. This study did not require accounting for dependence of cross-section and diffusion coefficients on boron concentration because of the underlying conservative assumption of the study. With no credit taken for injected boron or concentration increases as the result of boiling, only the base operating concentration is needed for calculation of kinetic parameters. The version of TRAC used for this study, however, has the ability to include boron concentration with fuel temperature and moderator density as an independent variable in the cross-section tables.

The fuel temperature ranged from 350 to 3200 °F, and the water density ranged from pure liquid

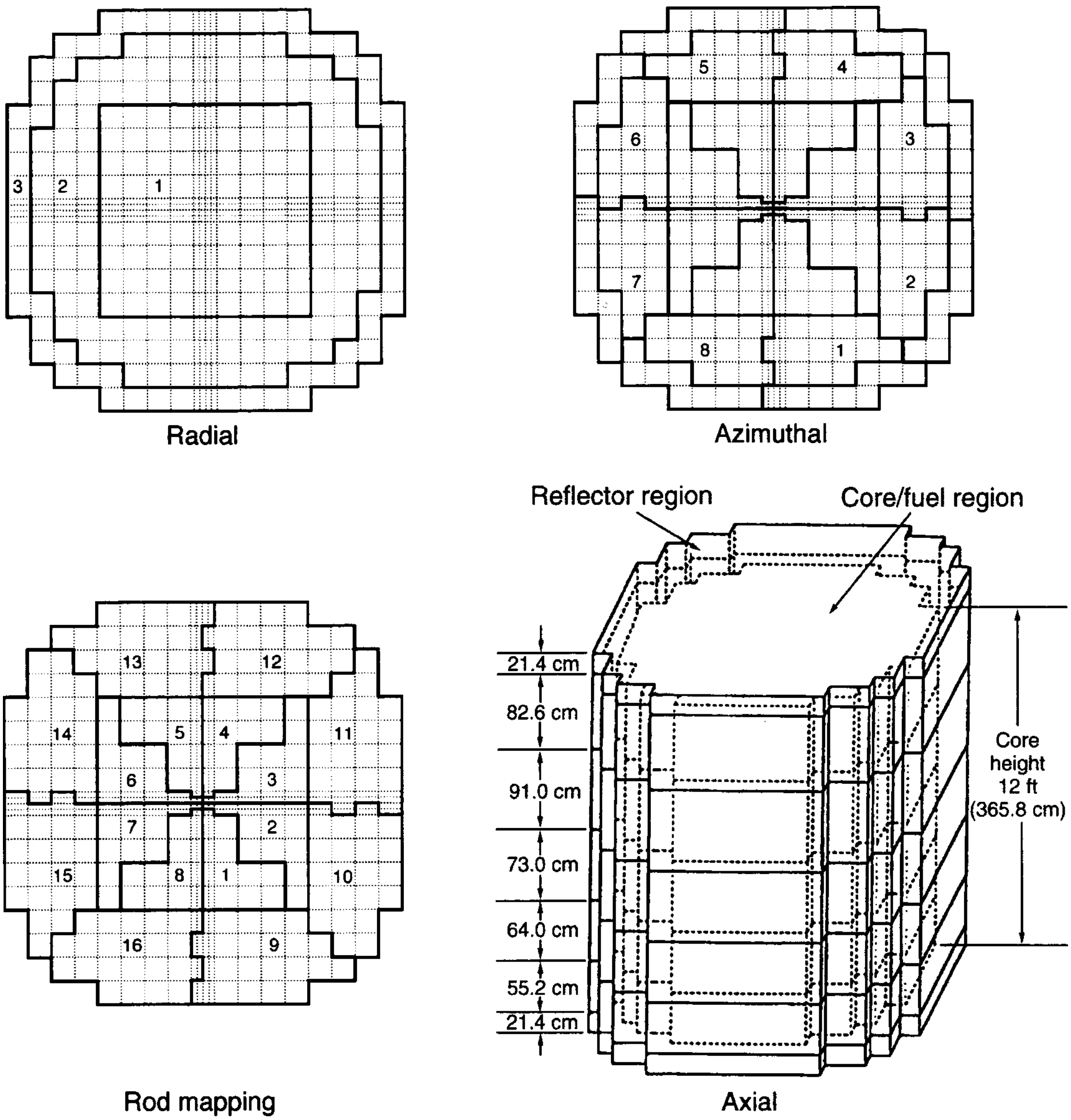

Fig. 1 3-D thermal–hydraulic–neutronic mapping.

at 220.0 °F to highly voided (void fraction = 0.6) fluid at 570.1 °F. In the case of an LBLOCA, where the core void fraction ranges from 0.0 to almost 1.0 at the end of blowdown, to decrease again after reflooding starts, void fraction is the most important parameter controlling coolant density. Void fractions larger than 0.6 were not considered in the table generation for two reasons: (1) the code selected to calculate the cross sections, PSU-LEOPARD,[10] could not handle accurately void fractions larger than 0.6; and (2) it was found that such large void fractions would yield very low moderation and could be neglected if compared with the moderating power of the high-density reflooding front flowing into the core. In fact, a run was made with interpolation of kinetics data to zero for void fractions equal to one with no appreciable differences in the results. As mentioned previously, the 3-D analysis can resolve the contribution of each neutronic node to the total neutron population in the core. Therefore it can account for the much greater influence that nodes coupled with low void fraction cells will have in the overall core neutronic behavior if compared with those nodes coupled to high void fraction regions. For nodes with void fractions higher than 0.6, the values of the cross sections at 0.6 were used. Regarding the possibility of whole core recriticality, the use of overpredicted moderation in the upper core (with expected void fractions larger than 0.6) will yield a more reactive core, which will result in a more severe transient.

The fuel elements were reduced to the unit cell scheme employed by PSU-LEOPARD, and the runs were extended up to xenon equilibrium burnup for a soluble boron equilibrium xenon concentration of 780 ppm. In addition, because the core model represented a first load core, burnable poison (BP) rods were present in some of the fuel assemblies. PSU-LEOPARD is not able to include discrete highly absorbing rods into the fuel assembly model it uses. Therefore the effect of the BP rods was accounted for by including an equivalent boron volume fraction in the clad region that would yield the same number density as that in the actual BP rods for the corresponding fuel assembly. This value was further modified by a factor that decreased the boron concentration to simulate the self-shielding effect on the neutron flux that a highly absorbing material produces. The same value was used for all fuel assemblies. The final value for this parameter was obtained by running several TRAC steady-state runs based on cross-section tables generated with different self-shielding factors until k_{eff} was equal to one.

Two different neutronic core models, heterogeneous and homogeneous core, were developed. The homogeneous model consisted of only one type of fuel assembly, representing a full core average fuel assembly and one type of reflector assembly. The results from this core were used as a guide and starting point for the more detailed heterogeneous core, which contained 11 different fuel assemblies and 1 type of reflector assembly. The 11 fuel assemblies represented the 3 different enrichments described previously together with different BP concentrations. This last characteristic was homogenized throughout the core because the NEM implementation could not converge to a solution without an excessively fine mesh when a highly heterogeneous core was used (i.e., assemblies containing BPs placed next to assemblies without BPs). This is related to the fact that the NEM method is based on diffusion theory. As is well known, in the vicinity of highly absorbing regions, diffusion theory does not yield accurate values of the neutron fluxes and currents. The version of NEM used in this study lacks the mitigating effects of discontinuity factors and applies a partial current formulation that is not particularly robust in the presence of abrupt changes in cross sections.

The homogenization of BPs is a conservative assumption from the point of view of the maximum clad and fuel temperature reached by the fuel rods. The BP rods are located in the center and in the highest enrichment fuel assemblies (e.g., the 2.7 wt % assemblies closest to the center of the core have the highest BP content). The homogenization process reduces the boron poison concentration in such assemblies, resulting in a higher power being predicted than the one that would actually exist if the poison were heterogeneously distributed. The power that those assemblies transfer to the ROD components coupled to them (the assemblies correspond actually to NEM nodes) is also higher, and this increases maximum values predicted for the clad and fuel centerline temperatures.

LBLOCA Simulation

The study was initiated with a steady-state run. To obtain convergence with the heterogeneous core, the steady-state run started with the homogeneous core model. This strategy allowed a smooth transition from the initial guesses for the thermal–hydraulic variables to their steady-state values without causing numerical difficulties for the neutronics calculation. After reaching a satisfactory convergence in the system parameters, the homogeneous core was replaced by the heterogeneous one, and the run was extended further until the convergence criteria were met (variations of <0.1% within a time step). The main steady-state system parameters are displayed in Table 1. They correspond to the actual plant values observed in a typical U.S./Japan PWR plant.

On the basis of the system steady-state configuration, the transient run was started with a complete double-ended (200%) guillotine break located in the largest pipe of the reactor cooling system (RCS). The break was simulated in the cold leg of loop 3, about 2.70 m from the end of the cold-leg nozzle (between the vessel injection nozzle and the ECCS injection point). The pressure history for the containment was described by pressure vs. time tables.

The system actions after such an event takes place were according to the standard response of automatic safety systems. The pumps were tripped by the pressurizer low-pressure signal. The SG secondary sides were isolated by closing the feedwater fills and the

Table 1 Main System Steady-State Parameters

Parameter	Value
Core power, MW	3 315
K_{eff}	0.9906
Pressurizer pressure, MPa	15.720
RCS loop flow, kg/s	4 637.0
Reactor flow, kg/s	18 550.0
Core flow, kg/s	17 710.0
Core bypass flow, %	4.5
Core initial temperature, K	551.4
Core outlet temperature, K	586.0
SG secondary flow, kg/s	449.3
SG steam pressure, MPa	4.964
SG steam temperature, K	536.6
SG feedwater temperature, K	493.4
Core average linear power, kW/m	22.42

steam outlet valves connected to the secondary sides. The injection of emergency cooling water was initiated by the accumulator's check valves opening on the basis of low-pressure readings in the cold leg (4.23 MPa). The ECCS (HPIS/LPIS), modeled as pressure-dependent fills, was tripped at 12.5 MPa in the cold legs with a 25-s delay to account for the inertia of fluid and pumps in the ECCS.

Regarding the core power evolution, three major conservative assumptions were made. First, the steady-state power was set to 102% of the nominal power (that is, 3315 MW); second, no SCRAM was activated after the LBLOCA took place; and third, no credit was given to the strong negative effect that the highly borated water from the accumulators and ECCS has in the core reactivity. Therefore the core power was mainly dependent on the moderation capacity of the coolant present in the core (liquid volume fraction) and on the Doppler feedback resulting from the increase of the fuel-rod temperature. Finally, the decay heat was calculated by the standard TRAC decay heat model: ANS 79 (Ref. 11).

The transient run was extended up to 400 s, when a certain steady state in the main system parameters was observed for more than 100 s.

RESULTS AND ANALYSIS

The transient begins with the double-ended guillotine break in loop 3 at time 0.0 s. Following this event, the pressure in the reactor vessel (see Fig. 2) decreases rapidly. This period, known as early blowdown, reduces the primary-side pressure to 6 MPa in about 8 s. Because of the fast vessel inventory depletion

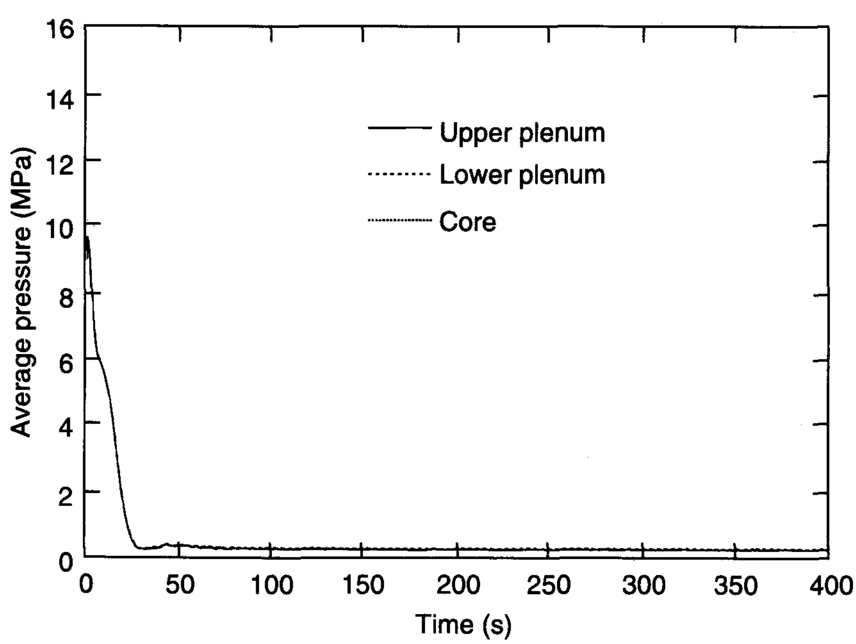

Fig. 2 Average vessel pressure.

through the break, the core water mass is suddenly reduced (see Fig. 3), and at 2.3 s the core is almost empty (void fraction 0.99 = liquid volume fraction of 0.01). As a result, the volume average fuel-rod temperature, a measure of the efficiency of rod to coolant heat transfer, rises sharply and reaches a first maximum around 1020.0 K (see Fig. 4), which is limited because of a partial core refill following the initial inventory depletion. This refill is produced by the inertia of the fluid in the cold legs and downcomer and by the pump coastdown. Figure 5 shows that the maximum clad temperature is also below 1000 K at the time of the first temperature peak.

The pumps, tripped at 3.5 s because of a low primary-side pressure signal, keep on pumping coolant into the cold legs until their stored inertia vanishes. After the core is half-filled at 5.5 s, its liquid inventory is reduced again because the reactor coolant is being lost through the break, and there is no other coolant source. By 30 s, the core is empty, and the clad temperature reaches a second lower peak of 950 K at 32.6 s. Both peaks are well below the limit value of 1475 K. This process takes place during the second part of the blowdown, when the pressure decreases more slowly (see Fig. 2) and the accumulators start the coolant injection into the cold legs at 13.5 s. The immediate effect is the rise of downcomer liquid fraction as it fills up with the water from the accumulators. By 24 s, the accumulators have discharged most of their contents, and the amount of coolant being injected into the downcomer drops abruptly. The downcomer liquid fraction is reduced sharply as the liquid flows into the core, whose liquid content raises again (see Fig. 3). Following this recovery of core inventory, the ECCS injection is initiated at 28.6 s. The surge of water from the ECCS into the vessel fills up the lower plenum at 30 s, and the reflood of the core begins. The lower core region is quenched at 31 s, and the reflood is completed near 175 s when the quench front reaches the top of the fuel rods (see Fig. 6). At this point the reactor pressure has stabilized at about 0.30 MPa, in equilibrium with the containment pressure.

During the reflooding process the reactor power remains low as shown in Fig. 7. The rod temperatures (Figs. 5 and 6) show no sign of the telltale increase that would be expected in an eventual return to whole core criticality, especially in the case of the centerline values. The core and downcomer liquid inventories grow slowly because of the LPSI. The total mass injected in the core is shown in Fig. 8. The strong mean flow reflected by the slope of these curves provides enough cooling to keep clad and fuel temperatures low.

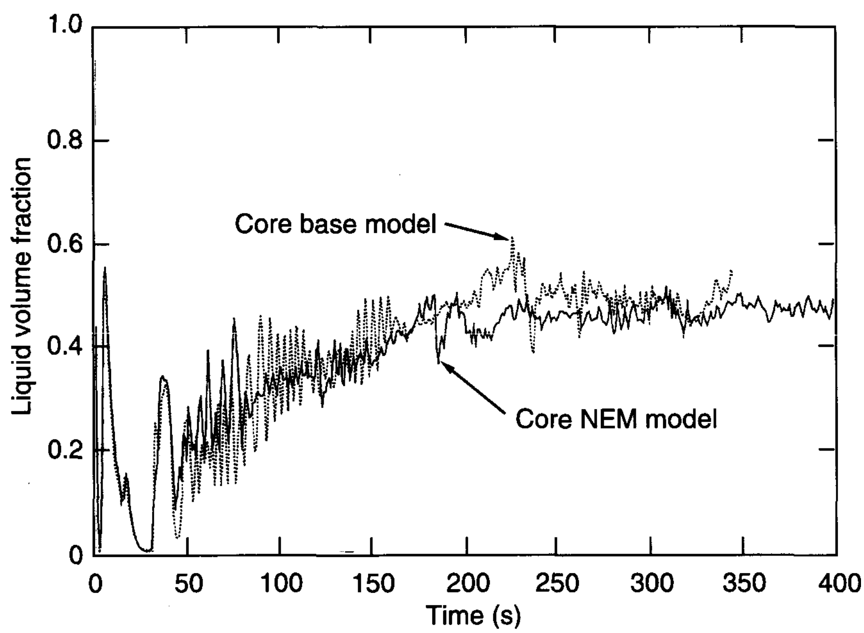

Fig. 3 Core liquid volume fraction.

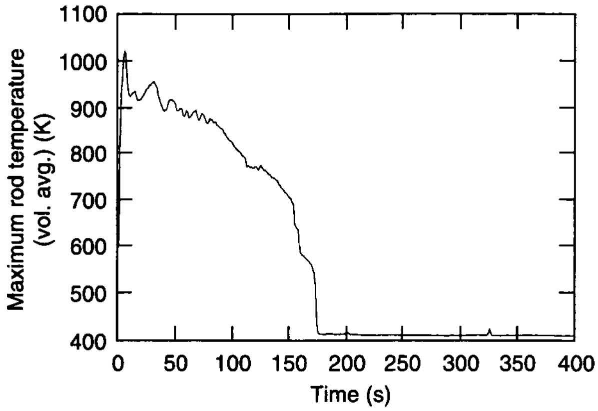

Fig. 4 Maximum average rod temperature.

The most relevant result is the stable state that the core reaches after 200 s. Figure 2 shows the stabilization of the liquid content in the vessel, with the curves leveling off at about 200 s. According to Fig. 8, liquid water is being injected into the vessel during the entire transient. Therefore the stabilization of the core liquid fraction around 0.45 is a sign of the evaporation of part of the incoming liquid as the power generated in the lower core settles into a slowly decreasing trend at a level about 5% larger than that expected from decay heat alone (see Fig. 9).

Figure 10 shows the normalized axial distribution of fission power at several times in the transient. One impact of this higher power level relative to the decay power can be seen in the centerline temperature plots shown in Fig. 5. With higher amounts of cold liquid in the lower core, one would expect lower temperatures in that region; however, the fuel centerline temperatures in the lower core regions (values at 3.09 and 4 m from bottom of the vessel) remain higher than in the middle (4.92 m) and upper (5.83 and 6.74 m) regions after the quenching has been completed.

The 3-D calculation describes the lower core based on the mostly liquid environment surrounding the fuel rods. Such conditions effectively moderate the neutron flux from the precursor decay (computed also by the model), which results in a higher fission rate and power production than that observed in the upper core, where the high void fraction cannot provide enough moderation to significantly increase the local fission rate. The total power being generated in the lower core boils off enough incoming liquid to keep the average core liquid fraction relatively stable and below the value necessary to drive the core into a critical state. A steady-state calculation with the conditions at 400 s showed a k_{eff} equal to 0.958068 when core liquid volume fraction is approximately 0.45. This value should be compared to additional studies with an artificial

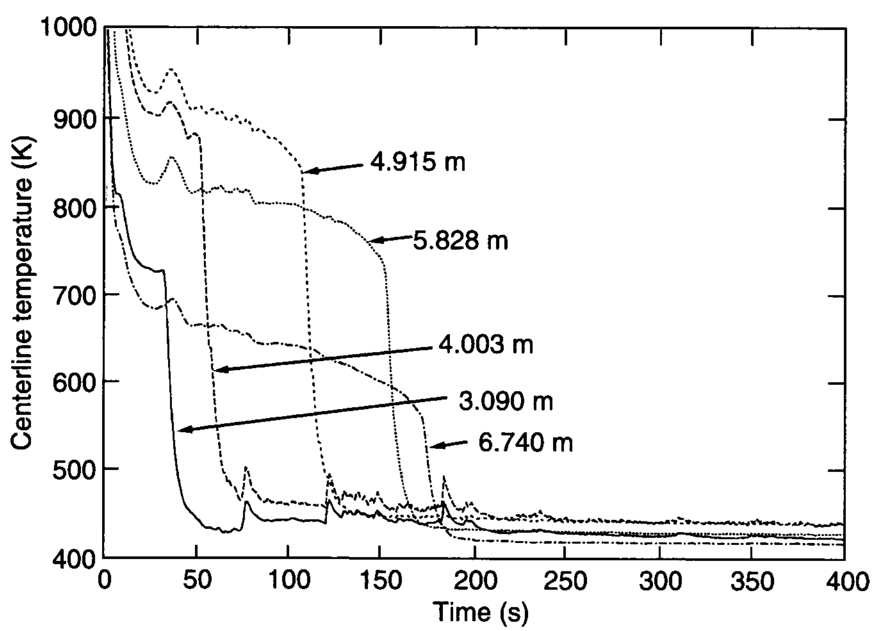

Fig. 5 Fuel centerline temperature.

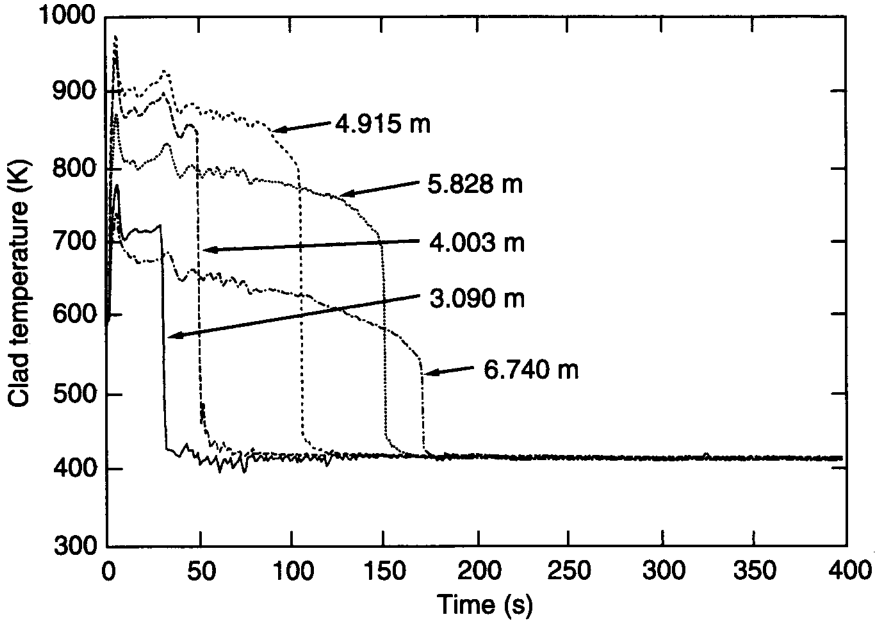

Fig. 6 Fuel rod clad temperature.

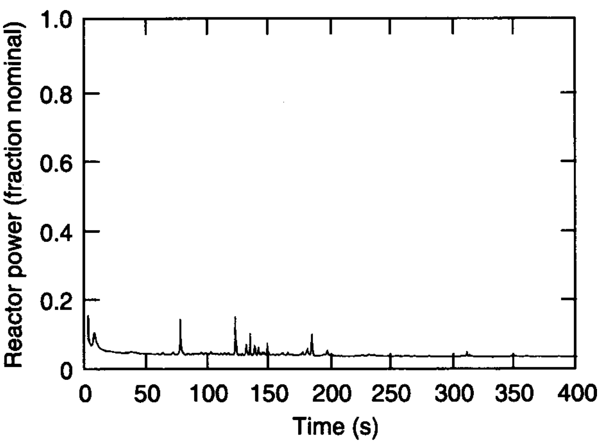

Fig. 7 Normalized reactor power (3315 MW = 1).

increase of core liquid inventory showing the minimum core liquid fraction for whole core recriticality to be about 0.62. Because conditions at 400 s represented the highest liquid inventory in the core after the initiation of the reflooding and the most important positive reactivity feedback in this analysis comes from the moderator density, the subcritical value for k_{eff} gives reasonable guarantees that the core remained subcritical during the whole transient. This assertion is supported by the comparison of the value of k_{eff} with a net $k_{eff,net}$ obtained according to the methodology proposed by Kim et al.[12] From a statistical analysis of the results presented in Refs. 5, 7, and 8 for several multidimensional benchmarks, the *corrected critical* $k_{eff,net}$ that would account for the bias of the NEM implementation used in the present study is 0.9697876. This value is larger than the subcritical k_{eff} computed for the system at 400 s and bounding because much of the actual bias in the method has been eliminated from this problem through the initial adjustments creating a critical steady-state core.

Although whole core recriticality does not seem to be likely under the conditions studied in this article, there still remains the possibility of local recriticality effects in the lower core regions. Figure 9 shows power spikes during the reflooding phase, which could be explained by such phenomenon. The source of any possible local recriticality, as mentioned previously, may be found in an increase in moderation of the delayed neutrons being produced in the lower core by the cold coolant. This would explain why the spikes are larger at the beginning and their peak values decay as the transient progresses (as neutron precursors decay). This process can well be attributed to local recriticality based on a balance of the neutron economy in the lower core neutronic nodes; however, it is not possible to be conclusive in this respect because the NEM method does not provide information about node k_{eff}. Nevertheless, as shown in the power plot, the local

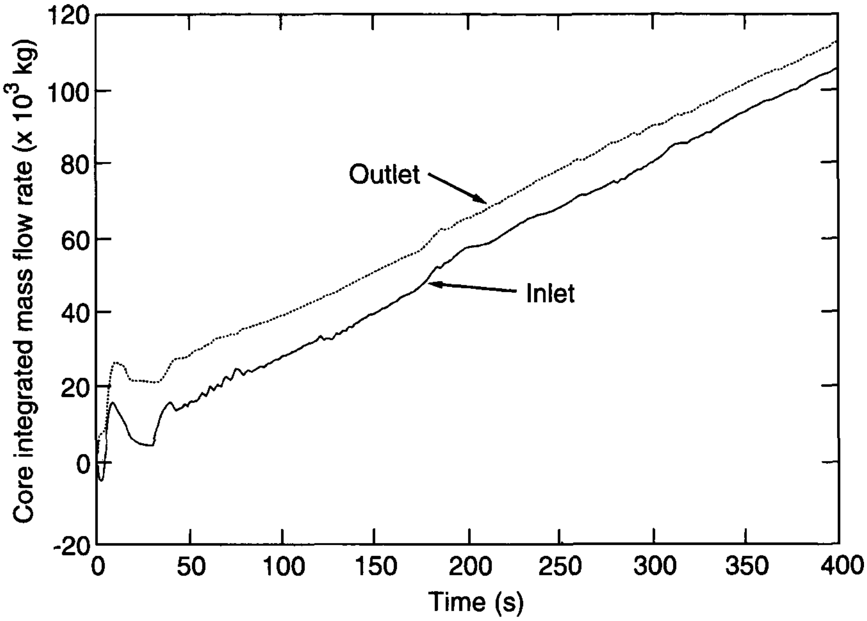

Fig. 8 Integrated mass flow rate through the core.

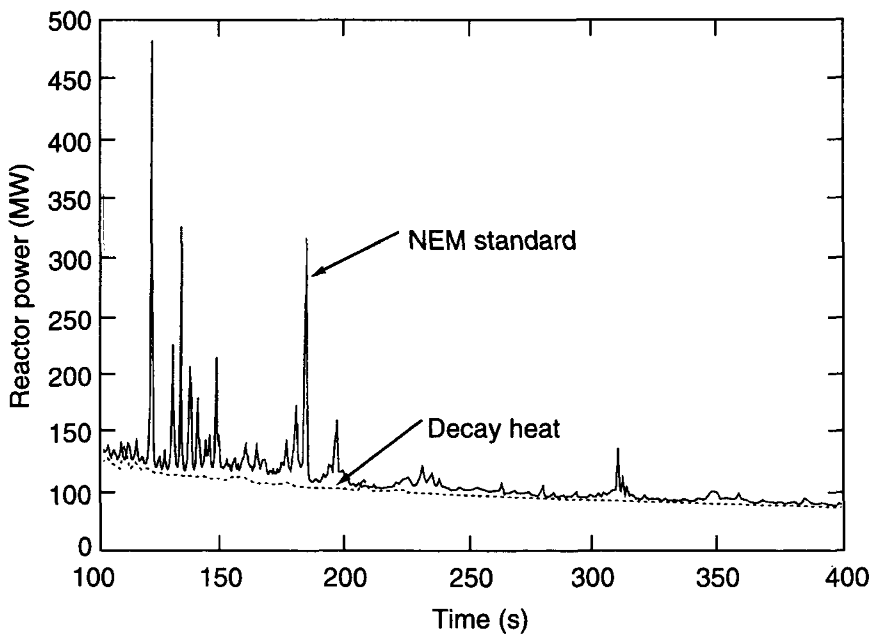

Fig. 9 Reactor power during post-LBLOCA reflood phase.

power spikes are a self-limiting process; the increase in power causes an increase in moderator temperature and oscillatory increases in void fraction (compare the behavior of the core liquid inventory coincidental with the power spikes). Conversely, the higher baseline power level, excluding spikes, is probably caused by the increase in moderation of the delayed neutrons, that, if enough to increase the number of fissions, is not enough to produce local recriticality until the right combination of fuel temperature, moderator

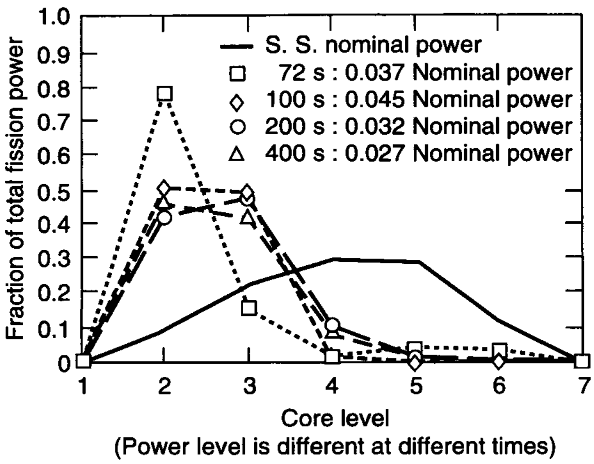

Fig. 10 Axial power profiles during the transient.

temperature, and void fraction (density) is locally reached. At that point, a sudden power spike appears. The extra power increases fuel temperature, void fraction, and moderator temperature; so enough negative reactivity is introduced to eliminate the local critical state until the next right combination is reached because cold coolant continues being injected in the core. Toward the end of the transient, the decay of neutron precursors is such that the amount of neutrons being released is not enough to result in local recriticality, and, although the baseline power remains larger than the decay heat power, the spikes have almost disappeared.

It is important to point out that the scale in Fig. 9 is such that the power oscillations can easily be resolved. The power spikes, however, are much smaller than rated power. As mentioned previosuly, further analysis showed a real return to power when the core liquid volume fraction reached a value of 0.62 by artificially collapsing the voids in the core (pressure wave). The resulting power spike observed was larger than nominal power and rapidly decayed as a result of thc negative fuel temperature and void coefficients of a PWR core.

SENSITIVITY CALCULATIONS

These results represent just one of several variations of transient assumptions and core models that were studied. We used a simpler homogeneous core to examine an LBLOCA with SCRAM and a transient without SCRAM. Runs were made with tracking of (but no feedback from) ECCS boron. In addition, a need was found to check the impact of the code's downcomer interfacial drag model.

Our first sensitivity test was to compare the results of this study against a base case matching this calculation in all respects except in that SCRAM was assumed at the beginning of the transient. This base case was originally run to check results from the current code against those from the last Los Alamos LBLOCA study (they matched reasonably well). We expected to see the additional fission power, reported in the previous section, result in a slightly lower late time core liquid fraction. Figure 3 shows this to generally be the case.

As indicated earlier, a second plant model was created with a homogeneous core. An LBLOCA without SCRAM was also run for this model and discussed by Tyler.[13] Results did not vary significantly from those reported here despite a significantly different radial core power distribution (maximum normalized assembly power was 2.0 for homogeneous vs. 1.4 for heterogeneous). In addition, that plant model included tracking of the boron injected from the ECCS. Figure 11 shows time histories of predicted boron calculations volume averaged over three different levels in the vessel. When we corrected for the numerical diffusion of the TRAC boron transport model, we found that boron levels of 2000 ppm were present by 35 s. The maximum boron concentrations were reached in the core within 45 s. Although the late time stability for the conservative assumption of no ECCS boron is reassuring, we expect even safer late time core behavior in a realistic LBLOCA without SCRAM, where credit is taken for boron injection. Late time core liquid inventories will be higher than those presented here, close to those obtained from calculations of LBLOCA with SCRAM, which will provide additional cooling with no chance of recriticality.

In reviewing our results, we realized that one set of standard LBLOCA "conservative" assumptions might not be conservative in this case. Because recriticality depends upon core liquid inventory, the assumption of minimum possible initial pressurizer and accumulator inventory might miss some important behavior. A study was performed with maximum initial inventory for these systems.[14] No major differences were observed in overall behavior.

One physical phenomenon that can have a major impact on liquid available to the core is downcomer bypass. Nithianandan[15] has noted that recent versions

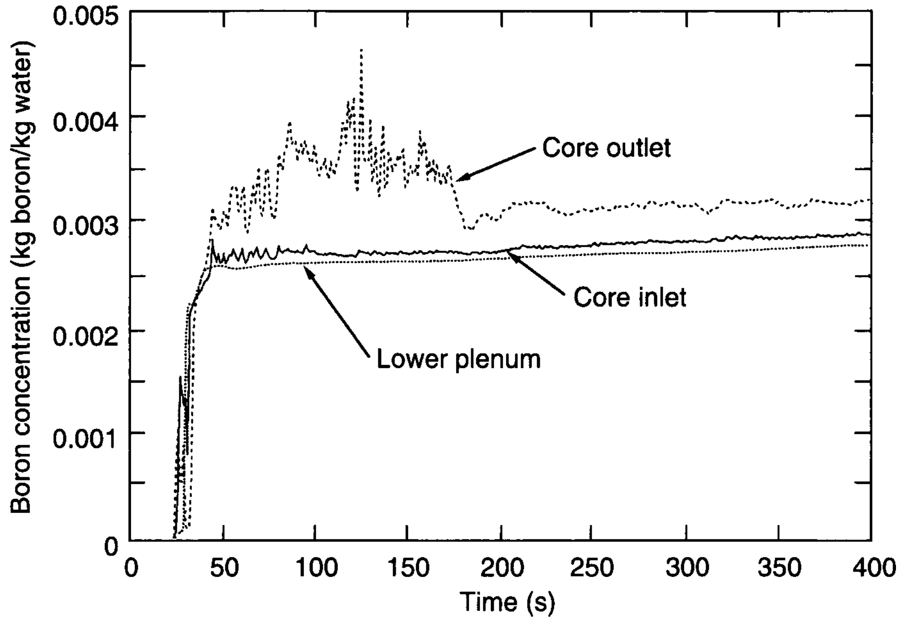

Fig. 11 Core and lower plenum boron concentration.

of TRAC-PF1/MOD2 underpredicted liquid delivery to the lower plenum in several relevant Upper Plenum Test Facility (UPTF) experiments. We confirmed this observation with tests of the version of the code used in this study. The source of the problem was identified as the interfacial drag model in the downcomer and lower plenum. Time did not permit us to develop a model that adequately matched a broad range of downcomer bypass experiments; however, we did obtain a model that matched or overpredicted delivery to the lower plenum for the UPTF tests. A 100-s run of the LBLOCA without SCRAM was made with this revised interfacial drag without major changes in overall system behavior.

It is worth noting that, although general system behavior, including core power, inventories, and pressures, did not vary widely during runs with several different interfacial drag models, details of the quench behavior did. Quench times changed significantly (>50 s) above the midplane, with changes as simple as different interpolations between flow regimes in the downcomer interfacial drag. This suggests that conclusions on the validity of core reflood models should be regarded with some caution when based on comparison with experiments not using forced flow to the rods.

CONCLUSIONS

The main objective of the analysis described in this article was to investigate the possible return to criticality during the reflooding phase of an LBLOCA. Scoping studies in which snapshots of the core thermal–hydraulic state are fed to a separate neutron kinetics code are too conservative because they do not properly model the self-limiting negative feedback that an eventual rise of reactor power would produce as a result of the increase in core void fraction. The tightly coupled neutron kinetics and thermal–hydraulic analysis package applied to this transient permitted us to observe the potential results of this feedback.

The analysis of the results has shown that a return to whole core criticality is not observed. The reactor power remains at a level slightly higher than decay heat (Fig. 9), and it is sufficient to effectively evaporate enough incoming liquid to keep the average core liquid fraction stable around 0.45. This value is, according to the power evolution plots (Figs. 7 and 9), below the threshold for achieving a whole core critical configuration. The stability of this situation is supported by the total integrated mass flow rates entering and exiting the core (Fig. 8). The continuing ECCS water injection maintains a constant flow through the core, which is adequate to ensure enough cooling of the fuel rods to prevent damage; however, the possibility of local recriticality has also been pointed out from the results observed in Fig. 9.

We think that more detailed analysis using the coupled methodology described in this article, especially with more refined thermal–hydraulic meshing, would be important to gain further insight on the dynamic response of PWR systems under such events by finely resolving local effects in the thermal–hydraulic–neutronic coupling. The ideal thermal–hydraulic noding scheme that would map each neutronic node with a corresponding thermal–hydraulic one is still in the future until more powerful computers can be used in transient analysis.

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Design Features

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Vulnerability of Multiple-Barrier Systems

By N. C. Lind[a]

Abstract: *"Vulnerability" is defined as the ratio of the probability of failure of a damaged system to the probability of failure of the undamaged system. This definition applies to all engineered systems and can be specialized to particular system types. Some disastrous failures (e.g., Chernobyl) have shown that systems can be highly vulnerable. "Defense in depth" is a powerful design principle, reducing vulnerability when the consequences of failure can be catastrophic. In the nuclear industry, defense in depth is widely used in radiation protection, reactor control, and shutdown systems. A multiple-barrier system is a simple example of a system that has defense in depth. The idea is that the system is not vulnerable. It cannot fail if one barrier fails because there is another to take its place. This idea is untenable in waste management, but a quantified vulnerability of a system can help owners, designers, and regulators decide how much defense in depth is desirable or enough. Many multiple-barrier systems can be modeled as systems of components physically in a series, each individually able to prevent failure. Components typically have bimodal distributions of the service time to failure, as illustrated by an example of application to a hypothetical nuclear fuel waste repository.*

The purpose of this article is to suggest and illustrate a quantitative measure of the "vulnerability" of a system. Any system should be able to sustain some damage without failure. Many failures and accidents can be ascribed, at least in part, to vulnerability or lack of "damage tolerance." Some examples from the nuclear industry occurred at the Brown's Ferry and Chernobyl power plants. Damage tolerance can usually be assured—but at a cost. If damage tolerance is to be optimized or regulated, then it must first be quantified. Quantifying vulnerability can help owners, designers, and regulators decide how much "defense in depth" is desirable, tolerable, or enough.

Vulnerability has been defined as the ratio of the probability of failure of the damaged system to the probability of failure of the undamaged system.[1] This definition generally applies to engineered systems and can be specialized to particular system types. Vulnerability and damage tolerance are reciprocal concepts. If a system is highly vulnerable, it has low damage tolerance and vice versa. It is convenient to define damage tolerance as the reciprocal of vulnerability.[1] Different concepts of vulnerability and damage tolerance have been proposed (e.g., Refs. 2 to 4).

Defense in depth is a powerful design principle; when consequences of failure can be catastrophic, it can be used to increase reliability and reduce vulnerability. In the nuclear industry, defense in depth is widely used in radiation protection, reactor control, and shutdown systems. A multiple-barrier system is a simple example of a system that has defense in depth. The idea is that the system is not vulnerable because it cannot fail if one barrier (or several) fails when another barrier remains to take its place. This deterministic idea is unworkable in the context of probabilistic analysis because it is explicitly admitted that all barriers can fail.

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Multiple-barrier systems may be modeled probabilistically as a system of components in parallel. This model is useful if the performance can be assessed in reliability terms, as when failure is not the expected behavior. Vulnerability of such systems can be analyzed directly.[1] In the design of waste containment projects, however, the issue is not if, but when, failure to satisfy the requirements will occur. Many multiple-barrier systems can be modeled instead as systems of components in a series. The failure mode is leakage, which is to be expected and sooner or later violates the constraints. The components are barriers that cannot prevent but can only delay transmission of the contaminants. (Transmission times can be so long as to result in effective prevention—that is what makes multiple-barrier systems useful.) The passage time is the sum of the passage times for all barriers, a random variable. Typically, barriers must be modeled by bimodal distributions of the time to failure because two distinct mechanisms can cause transmission. Other distribution models can be accommodated when necessary, as when a component has multiple failure mechanisms.

VULNERABILITY

For a precise definition of vulnerability, there should be a performance criterion that defines the failure of the system. Let $P(r,S)$ denote the probability of failure of the system for the prospective loading S, at a point r in space $\{r\}$, which for a dynamic system is the space of trajectories in system state space. Let r_0 denote a reference point in $\{r\}$; for a dynamic system this is the trajectory in $\{r\}$ that represents the behavior that is intended or expected. Denote a particular deteriorated or damaged state space trajectory by r_d. Then the conditional vulnerability V of the system in point r for prospective loading S is the ratio

$$V = V(r_d, S) = P(r_d, S)/P\{r_0, S) \qquad (1)$$

The vulnerability equals unity if the probability of failure is the same in the two states; otherwise it is generally higher. The system's life history takes it with probability $P(r_0)$ through one of a set of ordinary trajectories R_0. The alternative states, damaged or deteriorated, form another spectrum R_d for which the vulnerability is to be considered. Denote the expectation of the probability of failure over the sets R_0 and S by $P(R_0,S)$. Also denote the expectation of the probability of failure over the sets R_d and S by $P(R_d,S)$. Then the vulnerability V of the system is defined over the set of ordinary trajectories as

$$V = P(R_d, S)/P(R_0, S) \qquad (2)$$

For example, if something damages a system and increases the probability of failure threefold, then the associated vulnerability equals 3.

A MULTIPLE-BARRIER SYSTEM

A multiple-barrier system is modeled as a system of components in a series. An example is the concept for the disposal of Canada's nuclear fuel waste developed by Atomic Energy of Canada Limited (AECL):[5,6]

> Multiple barriers would protect humans and the natural environment from both radioactive and chemically toxic contaminants in the waste. These barriers would be the container; the waste form (spent-fuel pellets in zirconium alloy tubes, packed in fuel bundles with glass beads in the container); the buffer, backfill, and other vault seals; and the geosphere.[6]

Other nuclear fuel waste repositories are similar, differing not in principle but in details, particularly in materials.

Figure 1 is a schematic of a typical system. Groundwater seeps into the vault and clay buffer and then contacts and corrodes the containers. Groundwater also contacts and corrodes fuel sheaths and reaches the spent fuel. The contaminants are then released from the fuel and move through packing (e.g., glass beads), containers, and buffers. They then pass through a zone of a low-permeability geological medium (clay or rock, possibly via backfill in the vaults) and through the medium and its faults into the biosphere.

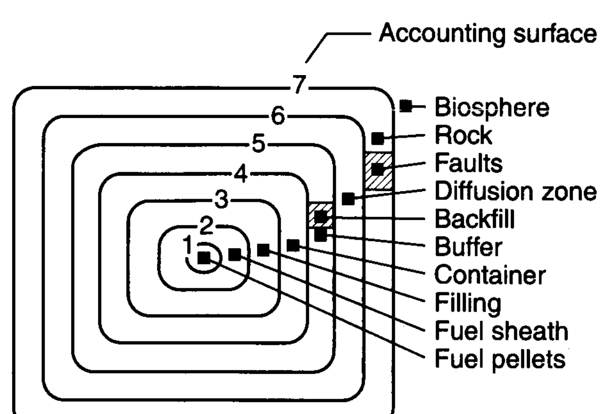

Fig. 1 Schematic of a multiple-barrier system.

Figure 2 is a schematic diagram of these processes. Each process i (i = 1, 2, ..., n; n = 8 in this case) is a transport of mass from one accounting surface to the next. The passage time elapsed in process i for a contaminant particle is denoted by T_i; it is a continuous random variable, in part because different particles follow different paths and have different speeds and in part because the parameters of transport are uncertain. Figure 3(a) is a schematic of the density function $f_T^0(t)$ of passage time T for a barrier.

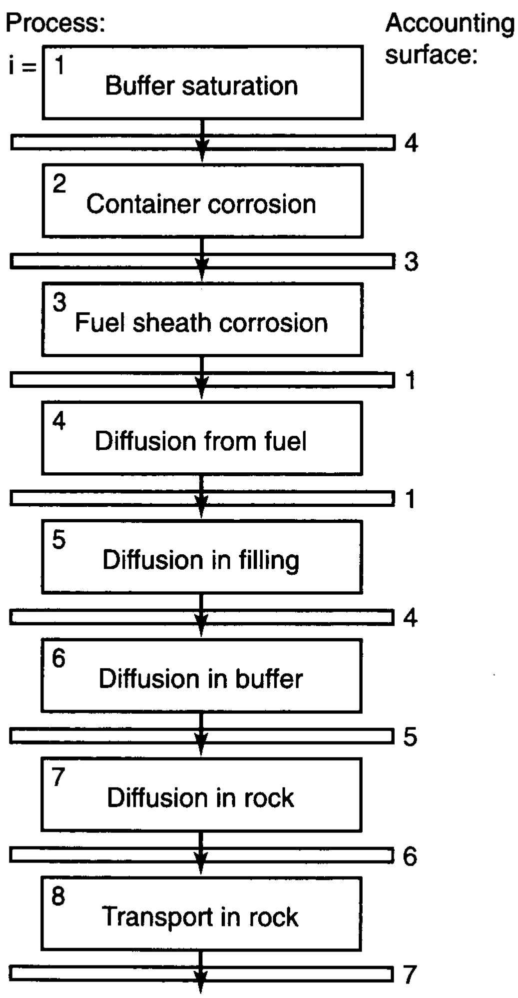

Fig. 2 Process block diagram for the multiple-barrier system in Fig. 1.

The passage time for a barrier would often be assigned a bimodal distribution as shown in Fig. 3(a) (for example, in the Ref. 5 high-level nuclear fuel waste repository concept, a few containers—on the order of 1% or less—will be defective at the time of placement, whereas others may fail early when the hydrostatic pressure and buffer swelling pressure build up). Most are expected by the AECL to fail much later by corrosion. A few fuel sheaths will be placed in broken condition, whereas others will break early upon pressurization; most should fail by corrosion. In particular, in the Canadian Deuterium–Natural Uranium Reactor (CANDU) fuel (UO_2 ceramic pellets) the majority of the contaminants are locked within the lattice of uranium and oxygen atoms and will be released only upon solution of the lattice; a small amount, depending on the species, is found on the grain boundaries, cracks, or gaps between pellets and sheaths and is released much faster. The buffer in the AECL concept is made of blocks of consolidated bentonite clay that is expected to swell considerably with water; however, it is

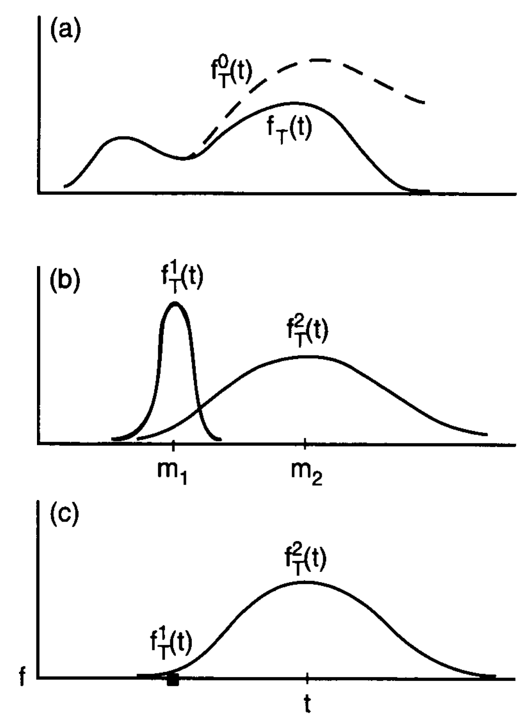

Fig. 3 (a) Typical bimodal distribution of mass transport (- - -) and net transport of an isotope in view of sorption and radioactive decay (—), (b) component distributions of a mixed Gaussian distribution, and (c) a fixed-point Gaussian distribution. [The $f_T(t)$'s are mass distribution density functions. m_1 and m_2 represent the means of Gaussian mixtures 1 and 2 and t represents time.]

prudent to assume that construction and inspection in a few cases leave a defective buffer. If the vault is situated in faultless rock, then passage of most contaminants will be by diffusion. Yet, stress from construction or tectonic forces may breach this barrier, whereas preexisting flaws may erroneously bypass all control procedures. Outside this diffusion zone the rock is assumed to have some flaws that provide relatively rapid convection of contaminants to the biosphere, but most may be assumed to be conveyed along with the general flow of groundwater in intact rock.

Different contaminants behave very differently in the system, so their passage must be analyzed individually. Many contaminants are radioactive; they transform by radioactive decay into other species. Some mass of each species is therefore lost by radioactive decay during passage. Each new species has a specific chemical behavior; some are sorbed or otherwise retained or delayed in the material of the barriers. These effects are shown schematically in Fig. 4. The result may be drawn into account by contaminant-specific adjustment of the passage time distribution density $f^0_T(t)$ into distribution density $f_T(t)$ [Fig. 3(a)], which may be interpreted as the mass distribution of the output from a barrier given a unit mass input at time $t = 0$.

The passage times T_i and T_j for two different processes i and j are normally statistically independent; however, it may be considered necessary to account for dependence, as in the AECL concept case, where clearly buffer saturation and diffusion in the buffer are correlated (Fig. 2, processes 1 and 6). Because the addition of passage times is commutative, any two correlated processes may be lumped into one process by convolution. The distribution of the lumped process may turn out to be multimodal.

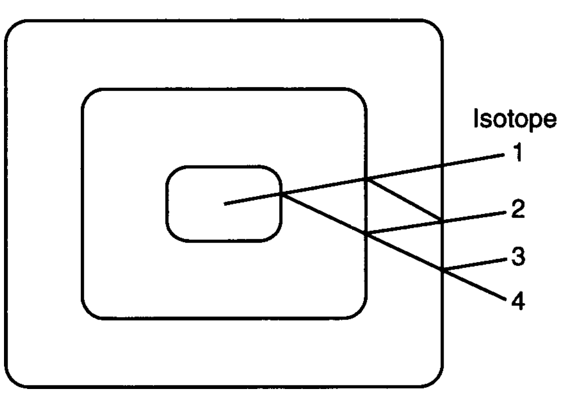

Fig. 4 Schematic of the transport of a decaying isotope.

SYSTEM ANALYSIS

Suppose that all processes are (or have been made) independent. Adding passage times T_i and T_j for two different processes i and j yields for their total passage time the distribution density

$$f_{i+j}(t) = d\left(\int_{-\infty}^{t} f_i(x)\int_{-\infty}^{x} f_j(y)dy\, dx\right)/dt \qquad (3)$$

in which x and y are dummy indices and subscript T has been suppressed. Repeated application of convolution [Eq. (3)] for all processes in arbitrary order yields the total passage time for the system. With the use of a convolution operator K, transforming f_i and f_j into f_{i+j}, Eq. 3 may be written symbolically as

$$f_{i+j} = Kf_i f_j \qquad (4)$$

It is easily shown that the convolution operator K is commutative, associative, and distributive. The distribution density of mass arrival time T_c in the biosphere for the contaminant may be written

$$f_c = K\,\Pi_1^n\, f_i \qquad (5)$$

If the mass inventory of this isotope is m at closure, time $t = 0$, then the mass released to the biosphere before time t equals $m' = mF_c(t)$, where $F_c(t)$ is the integral of $f_c(t)$ from time 0 to t.

Through the biosphere the contaminants follow a complex network of pathways to the recipients that eventually will receive a radiation dose or toxic dose and may suffer harm. Examples of pathways are {groundwater → stream → lake → sediment → benthic fish → predator fish → human} and {groundwater → soil → crop → meat → human}.

The conceivable harm includes cancer, genetic defects, and chronic poisoning and may be expressed collectively in a summary measure H. H may be the loss of life expectancy, quality-adjusted for health state, expressed in terms of days lost. H can be either individual or collective for a specified group of humans or other species.

The performance criterion relates in some way to the expected harm (for example, it could be specified that the expected harm to any person living downstream at any time before 1000 years after closure shall not exceed 2 weeks' loss of life in good health).

Performance criteria will generally be specified by regulatory authorities. A proponent will also have corporate responsibility and must set its own criteria. Professional and corporate ethics dictate other performance criteria, for which some guiding principles have been proposed.[7] A performance criterion need not involve expected harm explicitly (for example, the criterion might be expressed in terms of "the probability of catastrophic failure"). Nevertheless, harm is always the concern implicit in such criteria. The concept of harm as a function of release rate encompasses them all.

The expected harm is stochastic, a function $H(f_s)$ of the release from the biosphere. Let $f_{n+1}m'd\tau$ denote the probability of an increment dH to H at time $t + \tau$ as the result of an infinitesimal release m' to the biosphere. The biosphere system is thus formally treated as a component, labeled $n + 1$, extending the system analyzed, that transforms input $m'(t)$ into the expected harm H (this component is not subject to failure, of course):

$$H = H\,(mk\,\Pi_1^{n+1}\,f_i) \qquad (6)$$

By definition, the system's vulnerability V to failure of barrier i is the ratio of the probability of unacceptable harm with barrier i failed and all other barriers effective to the probability of unacceptable harm with all barriers effective:

$$V = H(mK\,\Pi_1^{n+1}\,f_j)\,/\,H(mK\,\Pi_1^{i-1}\,f_j\,\Pi_{i+1}^{n+1}\,f_j) \qquad (7)$$

The numerator in Eq. 7 reflects an assumed transmission time of zero for barrier i. More generally, vulnerability may be defined for any specified deviant ("rogue") component behavior. This is illustrated in the following example.

Numerical Analysis

If a mathematical model of the distributions f_i is available, the calculation of the vulnerability by Eq. 7 is simple in principle. The convolutions may be time-consuming, and exact evaluation may be impractical, but approximate calculations can be done by simulation (Monte Carlo or Latin Hypercube) or point-distribution methods.[8,9]

The analysis can be simplified when the process time distributions are Gaussian mixtures. Figure 3(b) shows a Gaussian mixture of two distributions. The distribution in Fig. 3(b) may be written as a weighted sum of two normal distributions $n(\ldots)$ or as a six-parameter mixture $M(\ldots)$,

$$\begin{aligned} f = f_1 + f_2 &= q_1 n(m_1, s_1^2) + q_2 n(m_2, s_2^2) \\ &= M(q_1 n, m_1, s_1, q_2, m_2, s_2) \end{aligned} \qquad (8)$$

The set may be considered as the union of two subsets, the rogues or defectives, and the normal components. Figure 3(c) shows a common special case in which a proportion q_1 is defective at time $t = m_1$ (m_1 would usually equal zero, but not necessarily).

If all distributions are Gaussian mixtures, then the convolutions in Eq. 7 will result in a Gaussian mixture of 2^{n+1} and 2^n normal distributions. Each of these is of the form of a factored normal distribution, $\Pi_{q_k} n(m, s^2)$, where m and s^2 are sums of means and variances in the distributions $M\{\ldots\}$. Further simplification is possible when the probability of a rogue q_1 is small in comparison with q_2 for all system components. This is illustrated in the following example.

Uncertainty

Some people believe that probability can be objective and demand that it should be. They feel that any probability that incorporates an element of uncertainty is an inferior basis for decision making; however, all probability that relates to the future of the real world is uncertain. It is sound policy to base important decisions on all available information, critically assessed. Information should be gathered when feasible until the value of the information one expects to gain is not worth the expected quality of choice in the decision.

There are two kinds of uncertainty in probabilistic system analysis: model uncertainty (that is, uncertainty about the relationships between quantities) and parameter uncertainty (that is, uncertainty about the parameters in these relations). Both kinds may be important in multiple-barrier systems. Model uncertainty is often given less than adequate treatment in scientific analysis, perhaps because it is difficult for a scientist to hold conflicting concepts of a process as "true" simultaneously. Model uncertainty can only be neutralized (with respect to its influence on the decisions that follow risk assessment) by making sure that a complete set of reasonably believable models is considered. Model uncertainty may be represented by a *model tree*, which gives a structured synopsis of the conflicting ways to model the phenomena. The model tree can be used to aggregate informed opinion into a compromise view of the risk.

Uncertainty in the parameters of the distributions f_i arises because they are estimated from statistical data, not always directly pertaining to the object, and they have a limited sample size. This uncertainty is often overlooked or ignored in probabilistic analysis from a classical point of view. Still parameter uncertainty can be accounted for within the quantitative risk analysis in a straightforward way [for example, suppose that a process (perhaps the way a contaminant is eliminated from a lake) is modeled by an exponential decay, $\exp(-t/A)$, where t is time and A is the attenuation time, a constant]. A is uncertain and may take on a finite set of possible values $\{A_1, A_2, ..., A_n\}$, some more believable than others. Each value A_i when applied in the model gives an associated harm, denoted H_i. Each value A_i is assigned a probability p_i expressing how likely it is thought to be. Parameter uncertainty is commonly judged by experts (that is, persons familiar with the relevant discipline). The expected value of the harm is the weighted sum $H = \Sigma p_i H_i$, which is then used in Eq. 7 to calculate the vulnerability. This vulnerability incorporates the views on the parameter uncertainty of the experts. If the spectrum of possible harm is (piecewise) continuous, the weighted sum is replaced by an integral.

In many cases the process models and the parameter distributions are mathematically tractable. They allow parameter uncertainty to be incorporated into the model by changing the uncertain process parameters $\{A\}$, taking expectation over the parameter distributions.[10] Because uncertainty is in itself very uncertain, it is often sufficiently accurate to use simple approximations. Hong,[9] using the point distributions of Rosenblueth,[8] has proposed a simple approximate method to account for parameter uncertainty.

NUMERICAL EXAMPLE

Consider a high-level nuclear fuel waste repository design as shown schematically in Fig. 1. For the example to be concrete and realistic, the system is structured like the Canadian concept described by AECL,[6] but the parameters are arbitrary and could represent any nuclear waste repository. The design is based on the philosophy of defense in depth. Some barriers are, however, more trustworthy than others. Quantitative risk analysis may rely on some of the barriers and disregard others.[5] Even when a reliable quantitative system analysis is presented, some people will be skeptical and may ask: "You say that the barriers cannot fail, but what if nevertheless one of the barriers fails?" Vulnerability addresses such questions.

The system vulnerability depends on the spectrum of failed states that are considered. In the present example, two such cases are studied: (a) any one barrier is completely ineffective, transmitting the contaminant instantaneously; and, more realistically, (b) any one barrier is a rogue, transmitting the contaminant faster than normal. Typically, physical components exhibit a high rate of failure twice during their anticipated lifetime, once shortly after being placed in service (defective or rogue behavior) and once nearer to the design life ("normal" deteriorating behavior).

Vulnerability depends also on the criteria of system failure and component failure. The system will be considered to fail if the effective radiation dose equivalent, to the most exposed person within 10 000 years after closure, exceeds a specified limit; for example, 1 mSv/yr has been prescribed in Canada by the Atomic Energy Control Board.[11] As it turns out, the rate of release of material to the biosphere increases monotonically over the first 10 000 years after closure. Also, after reaching the biosphere, the material is dispersed rapidly by convection in air and water. Therefore the most exposed person within 10 000 years after closure lives at the end of the period and immediately downstream from the effluent. Calculation of this exposure is complicated because of the transfer functions of many pathways and biological processes;[12] however, the biosphere input–output relation is fast and linear and has constant parameters. Therefore the contaminant concentrations from the intact and the damaged systems are proportional to the corresponding outputs from the geosphere; so the transfer functions of the biosphere and the receptor (human) cancel out in Eq. 7. Furthermore, the duration of the life of the most exposed person is short in comparison with the standard deviation of the output from the geosphere, so the environmental exposure and ingestion are approximately proportional to the density function $f_c(t)$ of the effluent of radioactivity for $t = 10\,000$ years. If the harm is stochastic, proportional to the exposure and ingestion, the vulnerability is calculated for both cases, namely (1) any barrier absent and (2) any barrier defective. This assumption is also examined in the discussion section.

Because of differences in chemical behavior and rate of radioactive transformation, each contaminant isotope must be analyzed separately and the expected harm summed over the set of all isotopes. Some isotopes are locked in the crystal lattice of the fuel and are released only as the pellets dissolve; others accumulate on the grain boundaries and in the gap between pellets and sheath. Depending on the pH potential, which varies over time and with location, some isotopes precipitate in the buffer and others travel freely by diffusion or convection. The example illustrates the transport of a soluble long-lived isotope, ^{129}I, which has a half-life of 15.7 million years and so does not decay appreciably during the 10 000 years of the criterion. The analysis for other isotopes is similar.

^{129}I is an important isotope (for example, in the estimate for the Canadian concept made by AECL,[5] the maximum radioactivity released to the biosphere during the first 100 000 years is about 100 times larger from ^{129}I than it is from ^{14}C and more than 10 000 times as great as that from any other isotope). It would seem that ^{129}I presents the greatest risk whether or not barriers have failed; so other isotopes could be neglected in comparison (but this may be wrong; ^{129}I may be quite innocuous, as shown in the discussion following). The distribution of each transport process time is assumed to be a mixture of two Gaussian distributions, characterized by their mean m (years), standard deviation s (years), and probability of occurrence q. The first component of each process is short, reflecting defective behavior, whereas the second component is of normal duration.

The distribution parameters for the barriers and the biosphere are given in Table 1. They are specific to the element (iodine). It is emphasized again that these values do not represent any real or contemplated system, although they are meant to be realistic. In particular, no attempt has been made to model the Canadian concept for a deep geologic repository, whose acceptability and possible site have not as yet been determined.[5,6]

There is a subtle difference in the distributions in Table 1. There are millions of fuel pellets and thousands of fuel elements, containers, and clay buffers. This means that realization of the rogues in processes 1 to 6 is practically certain and will occur in the proportion q listed in column (3) in Table 1. But there will be only one rock or sediment environment and only one biosphere, so the probabilities in column (3) for processes 7 to 9 reflect a chance taken. In decision making based on the mathematical expectation of outcomes, this difference is of no consequence.

The expected system behavior is calculated first (Table 2). Multiplying the probabilities from column (6) in Table 1 gives the probability that all processes proceed in the normal mode, q = 0.74 in column (5). Summation of the means and variances from columns (7) and (8) in Table 1, respectively, gives the mean and variance of this eventuality. The expected efflux from the biosphere in the normal mode at 10 000 years [Table 2, column (9)] is calculated as the density of a normal distribution with mean m = 32 350 years and variance $s^2 = 5\,780^2$ years^2 at x = 10 000 years (Ref. 10). The expected efflux, meaning the annual fraction of the inventory flowing out of the biosphere at 10 000 years, is 2.9×10^{-8} per year in the normal mode, shown in column (9) in Table 2.

To this normal-mode flux should be added the probability-weighted effluxes in the rogue modes. Each process i = 1 to 8 has a *complementary* process in

Table 1 Example—Process Time Distribution Parameters, ^{129}I

Distribution		First (rogue) component			Second component		
i (1)	Process (2)	q (3)	m (4)	s (5)	q (6)	m (7)	s (8)
1	Vault and buffer saturation	0.03	50	20	0.97	200	50
2	Breach of container	0.05	0	0	0.95	2 400	600
3	Breach of fuel sheath	0.05	0	0	0.95	500	200
4	Escape from fuel	0.04	1 000	500	0.96	5 000	2 000
5	Transport in filler	0	0	0	1	200	100
6	Transport in buffer	0.05	50	20	0.95	9 000	3 000
7	Diffusion in exclusion zone	0.02	500	200	0.98	5 000	2 000
8	Flow in rock	0.05	2 000	1 000	0.95	10 000	4 000
9	Biosphere pathways	0	0	0	1	50	20

Table 2 Example—Calculation of Annual Expected Efflux Rate of ^{129}I at 10 000 Years After Closure

			Parameters:	q	m	s	Efflux per year	
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)
Normal processes 1–9:			*	0.74	32 350	5 780		2.9×10^{-8}
Rogue processes:								
	Complementary process			First-order term				
i	q	m	s	q	m	s	Efflux per year	
1	0.77	32 150	5 780 *	0.02	32 200	5 780	9.9×10^{-10}	
2	0.78	29 950	5 749 *	0.04	29 950	5 749	6.6×10^{-9}	
3	0.78	31 850	5 777 *	0.04	31 850	5 777	2.1×10^{-9}	
4	0.77	27 350	5 423 *	0.03	28 350	5 446	7.8×10^{-9}	
5	0.74	32 150	5 780 *	0.00	32 150	5 780	0	
6	0.78	23 350	4 941 *	0.04	23 400	4 941	8.0×10^{-8}	
7	0.76	27 350	5 423 *	0.02	27 850	5 427	5.4×10^{-9}	
8	0.78	22 350	4 173 *	0.04	24 350	4 291	1.4×10^{-8}	
	Sum of rogues:						1.16×10^{-7}	1.16×10^{-7}
	Total efflux at t = 10 000 years:							1.45×10^{-7}

the set i = 1 to 9; this is the process obtained by deleting element i from the set of second components listed in Table 1. The parameters are calculated as just described for the normal system behavior by substituting q = 1, m = 0, and s = 0 in row i, columns (6) to (8). They are listed in columns (2) to (4) in Table 2. They serve as auxiliary quantities to calculate the probability density of each rogue mode; for example (see Table 1, row 1), consider that the vault and buffer saturate early, in the mean 50 years after closure instead of 200 years. Then q becomes (0.3)(0.77) = 0.2, the mean time of reaching the biosphere drops by 200–50 = 150 years, from 32 350 to 32 200 years, and the variance is reduced by 50^2 to 20^2 [which is not enough to give a noticeable reduction in the standard deviation as Table 1, column (7) shows]. The effluxes in column (8) of Table 2 are calculated as for the normal mode. The sum of these single-rogue effluxes is 1.45×10^{-7} per year as shown in column (9). Of course, it is possible that two or more processes proceed in the rogue mode, and these eventualities should be examined, taking correlations (common-cause) into account, but the combined probability of multiple misbehavior is very small. Neglecting such rogue behavior, the expected efflux of this isotope at 10 000 years for the expected system performance is 1.45×10^{-7} per year.

Next, the expected effluxes at 10 000 years for the cases of (a) an absent barrier or (b) a defective barrier are calculated. In case (a), one or more barriers are completely ineffective. What if one barrier were to fail, transmitting the contaminant instantaneously? Calculations for this case are analogous to those shown in Table 2, but the time is replaced by 0 for the failing process. This increases the efflux at 10 000 years by the factor listed for a few barriers in column (3) of Table 3.

Table 3 Example—Vulnerabilities

i (1)	Failing barrier process (2)	Case (a) (3)	Case (b) (4)
1	Fast vault and buffer saturation		1.0
2	Early breach of containers		1.6
3	Early breach of fuel sheaths		1.1
4	Rapid escape from fuel		2.0
5	Transport in container fill		1.0
6	Transport in buffer	26.3	10.7
7	Flow in exclusion zone	30.9	2.5
8	Flow in rock		2.5
7+8	Flow in rock (both zones)	158	19.4

What if two barriers fail together? The case of two barriers failing simultaneously, however unlikely, is a valid subject of conditional vulnerability and is analyzed in the same fashion. The last row of Table 3 illustrates this event. Both rock zones may fail—perhaps, if one is faulty, the other could likely be faulty, too. Yet, since one rock zone is assumed to transmit the isotope by diffusion while the other transmits by convection, the stochastic dependence may not be pronounced. It is neglected in the present analysis; but if there are adequate data, it may be taken into account by modifying the distributions in Table 1. Transport in geological media is complex and beyond the scope of this article.

In case (b), any one barrier is a rogue, transmitting the contaminant rapidly. Again, the calculations follow the pattern of Table 2, but the probabilities for the barrier in question in columns (3) and (6) of Table 1 are replaced by unity and zero, respectively. Column (4) in Table 3 shows the vulnerabilities if process failure is defined in this way.

DISCUSSION

Quantitative definition of vulnerability makes it possible to specify a minimum tolerable or allowable value in a code, standard, or regulation. It would also allow setting target values and acceptable values for good design of particular classes of systems. The cost of reducing vulnerability may be high, and resources may be better spent on other objectives. There may be an optimum balance between reliability and vulnerability of a system.

Notice that the initial inventory m cancels out in the vulnerability in the example. The reason is that it is a common factor of the harm function, assumed linear in the flux of contaminant from biosphere to recipient, as follows from the conventional assumption of linearity in the dose–harm relationship. The assumption would be valid for most isotopes but is likely inaccurate for ^{129}I. ^{129}I is only slightly radioactive; it is a slow beta emitter with a half-life of about 15.7 million years. The human thyroid gland concentrates iodine but can hold only a few milligrams, which imposes an absolute upper limit on the dose to the thyroid. The rest of the body maintains a much lower concentration of iodine in proportion to the concentration in the thyroid. This limits the probability of cancer at high exposure, so harm is not linearly related to efflux. Although dose–response linearity is a fundamental postulate in radiation protection and regulatory practice, it is seriously in doubt as a scientific hypothesis for risk analysis purposes for very low exposures. Limited by thyroid capacity, the radiation risk to an individual from ^{129}I may well be negligible. If the risk from other isotopes is smaller yet, the individual dose criterion of harm becomes less tenable as a surrogate for total harm to the population. Then the contaminant flux rate used in the example should be replaced by the total contaminant released, $F_c(t)$; the calculations of vulnerability would be similar.

Case (a) in the example casts some light on cases that may be unlikely but still of concern. A conditional vulnerability greater than 100 (if the geological barriers fail) may or may not be acceptable. It is certainly a signal that almost "all the eggs are in one basket."

Quantifying vulnerability as in case (b) in the example indicates which barriers are relatively important. Table 3 would suggest that the containers and the filler are unimportant. Such a conclusion should be tempered with consideration of possible alternatives; for example, long-lived containers (e.g., copper) or a different filler may be substituted, which would drastically reduce the probability of harm. The vulnerability would simultaneously increase and point to the importance of these barriers.

Distribution assumptions must be documented in detail and justified according to accepted protocol. In practice, it will not be adequate merely to assume that all distributions are a mixture of two Gaussian components. Several Gaussian distributions can be mixed for a better representation if justified by the data; the convolutions can be done in closed form as in the example. Generally, the calculations would require numerical convolution with a large computer.

Approximations as simple as in the example can be useful if done with caution. Multiple-sum processes tend toward a Gaussian distribution by the Central Limit Theorem. If the output can be modeled as Gaussian, it is not necessary to assume distribution type for the components. Also, errors in a distribution appear analogously in numerator and denominator of the vulnerability and so tend to cancel out.

CONCLUSIONS

A quantitative measure of vulnerability can be useful in the assessment of the adequacy of a proposed or existing system. The measure must be probabilistic because deterministic measures fail to capture an essential feature of vulnerability: the reduction in reliability of a system that is damaged but has not failed.

The vulnerability of a system is a function of the state or trajectory of the system and the loading. To calculate vulnerability, the probability of failure is compared in two sets of states: the reference state (null, original, pristine, or initial) and the alternative state (rogue, damaged, deteriorated, or modified), which form a spectrum. Conditional vulnerability is defined for particular alternative states and prospective input as the ratio of the failure probability in that state to the failure probability in the reference state; for a spectrum of alternative states and prospective inputs, vulnerability is calculated by the total probability rule. The example, though simplified, shows that the vulnerability concept is objective and can be calculated.

ACKNOWLEDGMENTS

The Natural Sciences and Engineering Research Council of Canada provided some financial support for this study. The author is indebted to Don Wiles for the observation that the long half-life and low thyroid capacity limits the possible harm to an individual from ^{129}I and may invalidate the assumptions in the example. Thanks are also due to the reviewers of *Nuclear Safety* who provided many useful suggestions.

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Environmental Effects

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A Study of Wet Catalytic Oxidation of Radioactive Spent Ion Exchange Resin by Hydrogen Peroxide

By Xingchao Jian,[a] Tianbao Wu,[a] and Guichun Yun[a]

Abstract: *The decomposition behavior of cationic, anionic, and mixed ion-exchange resins was investigated in the* H_2O_2*–*Ni^{2+}/Cu^{2+}*,* H_2O_2*–*Mn^{2+}/Cu^{2+}*,* H_2O_2*–*Fe^{2+}*,* H_2O_2*–*Cu^{2+}*, and* H_2O_2*–*Fe^{2+}/Cu^{2+} *systems for volume reduction and improvement in the capacity of the cemented product. The effects on reaction processes and the consequences of many other factors were analyzed. No radioactivity was detected in the off-gas. The cementation process of encapsulation of the concentrated decomposition residue could produce qualified cemented products with excellent properties for long-term storage in a volume-reduced state.*

Radioactive spent ion-exchange resin (IER) is one of the main kinds of solid wastes produced by nuclear installations. Direct solidification of spent IER by cementation is currently the main immobilization process. The cost of transportation and ultimate disposal of spent IER increases considerably when directly solidified because the volume increases more than 80%. Therefore volume reduction technology has been studied. The research results reported by the U.S. Electric Power Research Institute (EPRI) showed that all the volume reduction processes had clear economical effects compared with the nonvolume reduction processes.[1] Spent IER volume reduction processes, such as incineration, pyrolysis, acid degradation, and high-temperature wet oxidation, have some disadvantages, such as the need for high operating temperatures and the production of radioactive off-gases. It is well known that the decomposition of hydrogen peroxide by catalysis with Fe^{2+} is a chain-free radical reaction that yields highly reactive hydroxyl radicals. The process has been widely investigated as a prospective option for the treatment of spent resin. Catalytic low-temperature wet oxidation has evident advantages in radioactive waste treatment because of its moderate operating conditions and sufficient volume reduction effect. Calculated results by B. G. Place indicate that ultimate disposal by hydrogen peroxide oxidation of spent IER costs about 50% less than disposal by direct solidification.[2]

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A BRIEF INTRODUCTION TO THE DECOMPOSITION PROCESS

The resins used in this study were cross-linked polystyrene strong acidic and basic resins. The structures of such IERs are shown in Fig. 1. The mesh structure of the IERs made them stable.

Wet catalytic oxidation of IER is a chain reaction initiated by the hydroxyl radical (HO•). The reaction between hydrogen peroxide and ferrous sulfate is the typical free radical reaction discovered by Fenton in 1894 and described by Harber and Weiss as follows:[3]

$$Fe^{2+} + H_2O_2 \rightarrow Fe^{3+} + OH + HO\bullet \quad (1)$$

$$HO\bullet + H_2O_2 \rightarrow H_2O + HOO\bullet \quad (2)$$

$$HOO\bullet + H_2O_2 \rightarrow O_2 + H_2O + HO\bullet \quad (3)$$

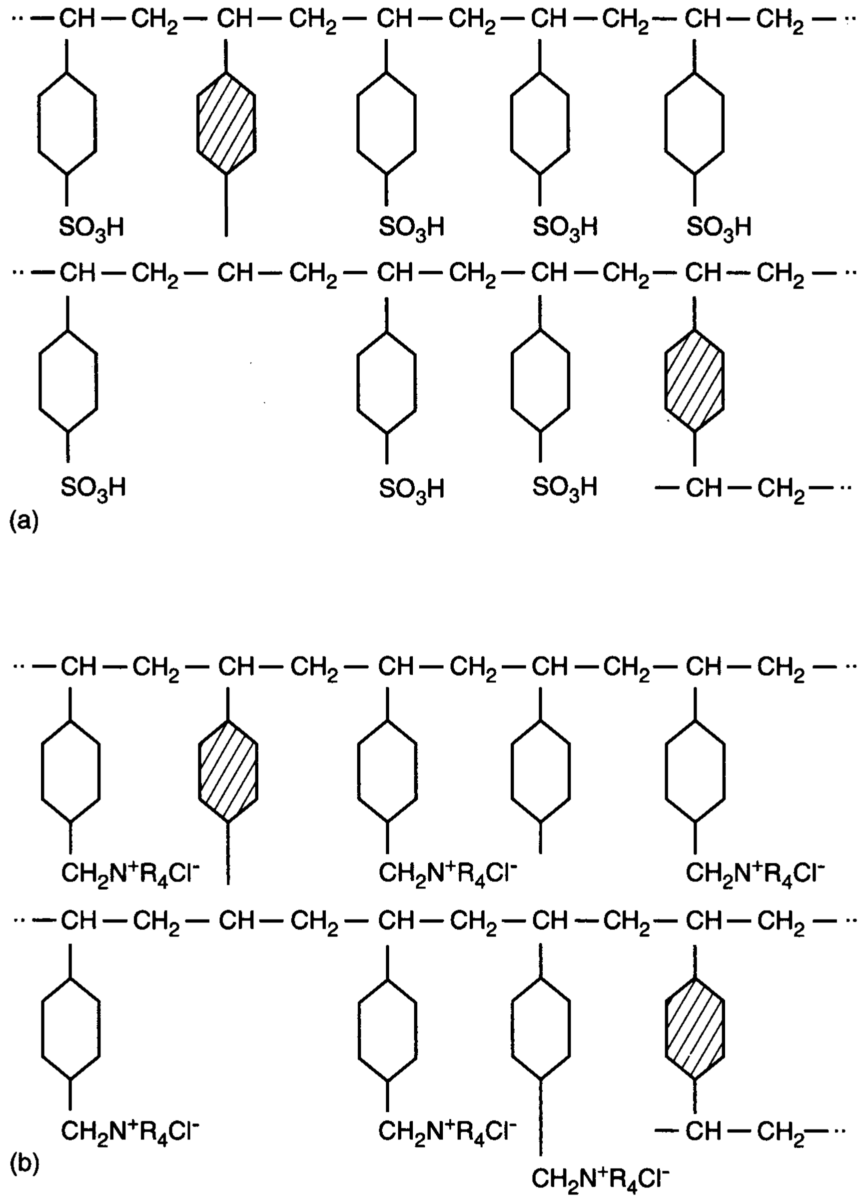

Fig. 1(a) Structures of strong acid polystyrene cation resin and (b) structures of strong base polystyrene anion resin. [Black benzene means it is connected by two carbon chains (this kind of connection makes the resin into a mesh). Blank benzene means it is connected with one carbon chain.]

$$HO\bullet + Fe^{2+} \rightarrow Fe^{3+} + OH^- \quad (4)$$

$$Fe^{3+} + H_2O_2 \rightarrow H^+ + Fe^{2+} + HOO\bullet \quad (5)$$

$$Fe^{3+} + HOO\bullet \rightarrow Fe^{2+} + O_2 + H^+ \quad (6)$$

The hydroxyl radical, which is a highly reactive radical, can react with organic substances either by hydrogen abstraction or by addition to an unsaturated hydrocarbon.[4] The oxidation of organic substrates by the hydroxyl radical, which involve reactions with organic free radicals (R•), can be represented as the following reactions:

$$RH + HO\bullet \rightarrow H_2O + R\bullet \quad (7a)$$

$$R{-}C = C{-} + HO\bullet \rightarrow R{-}C{-}C{-}OH \quad (7b)$$

$$R\bullet + O_2 \rightarrow ROO\bullet \quad (8)$$

$$ROO\bullet \rightarrow R' + CO_2 \quad (9)$$

$$2R\bullet \rightarrow R{-}R \quad (10)$$

These organic free radicals (R•), which are involved in the preceding equations, can be oxidized by high valence ions such as Fe^{3+} and Cu^{2+}, as shown in the following reactions:[5]

$$R\bullet + Fe^{3+} \rightarrow R^+ + Fe^{2+} \quad (11)$$

$$R\bullet + Cu^{2+} \rightarrow R^+ + Cu^+ \quad (12)$$

In these reactions, Cu^{2+} is a more effective oxidant than Fe^{3+} for simple alkyl radicals, and the produced Cu^+ (in Eq. 12) can increase the concentration of Fe^{2+} because of the existence of the following reaction:[5]

$$Cu^+ + Fe^{3+} \rightarrow Fe^{2+} + Cu^{2+} \quad (13)$$

During the dissolution of IER, the polymer is dissolved gradually by hydroxyl radicals, as described in the following reactions:

$$HO\bullet + IER \rightarrow \text{linear polystyrene (soluble)} + CO_2 + SO_4^{2-}/NH_4^+ + H_2O \quad (14)$$

$$\text{Linear polystyrene} + HO\bullet \rightarrow \text{simple organic substrates} + CO_2 + H_2O \quad (15)$$

The results of the resin decomposition are as follows:

$$C_8H_8SO_3 + 20H_2O_2 \rightarrow 8CO_2 + 23H_2O + H_2SO_4 \quad (16)$$

$$C_{12}H_{19}NO + 31H_2O_2 \rightarrow 12CO_2 + NH_4OH + 38H_2O \quad (17)$$

$$C_{10}H_{10} + 25H_2O_2 \rightarrow 10CO_2 + 30H_2O \quad (18)$$

$$C_8H_8 + H_2O_2 \rightarrow 8CO_2 + 24H_2O \quad (19)$$

Equations 16 and 17 represent the dissolution reactions of functional groups, while Eqs. 18 and 19 stand for the reactions of the cross-linking agent ($C_{10}H_{10}$) and the styrene unit (C_8H_8), respectively.

EXPERIMENTS AND RESULTS

Material and Equipment

The strong acidic and basic polystyrene resins (in granular form) chosen for this study (labeled as 732 and 711, respectively) were made in China and contained 45% water. 30% (vol.) hydrogen peroxide (C.P.) was used as an oxidant and 0.1*M* $Ni(NO_3)_2$, $MnSO_4$, $Cu(NO_3)_2$, and $FeSO_4$ (A.R.) solutions were used as catalysts. Moreover, NaOH (5%, C.P.) and antifoaming agent (XP-1, C.P.) and other instruments were needed. Normal Portland cement (labeled 525) made by the Sichuan JiangYou cement factory and sulfate resistance cement (labeled 525) made by the TianJin special cement factory were chosen for the cementation of the anion and cation resins, respectively.

All decomposition experiments were conducted in a 500-mL four-necked glass flask equipped with a water condenser, a mechanical agitator, and a thermometer. The flask was heated by an electric plate. A sketch of the laboratory resin oxidation system is shown in Fig. 2.

Process Description

A series of H_2O_2–Mn^{2+}/Cu^{2+}, H_2O_2–Ni^{2+}/Cu^{2+}, H_2O_2–Fe^{2+}/Cu^{2+}, H_2O_2–Cu^{2+}, and H_2O_2–Mn^{2+}/Fe^{2+} decomposition systems was studied. Fifteen grams of wet resins and a portion of the catalyst solution (about one-fourth of the total) were added to the flask first. When the temperature of the reaction system reached 90° C, 30 vol% hydrogen peroxide and the remaining catalyst were added to the flask, and the reaction time began to be recorded. Hydrogen peroxide was added at a rate of about 30 mL/min, and the catalyst was added at about 15 mL/min (total amount of 50 mL) consecutively. The total reaction time was 2.5 to 3.5 h.

(1) Anion resin could be decomposed in one of four systems: Ni^{2+}/Cu^{2+}, Mn^{2+}/Cu^{2+}, Fe^{2+}/Cu^{2+}, and Cu^{2+} systems. These four systems had almost the same reaction effect: at the beginning of the reaction, the dissolution of resins was violent, and a large amount of CO_2 was emitted. An hour later, all the resin beads turned into black liquid. Antifoaming agents were added, and a yellowish solid substance appeared and increased gradually in 1.5 to 2.0 h. The antifoaming agent clearly assisted these solids in binding together. These yellow solids floated on the surface of the liquid and bonded together when the foaming process ended in about 2.5 h. The amount of the produced CO_2 decreased,

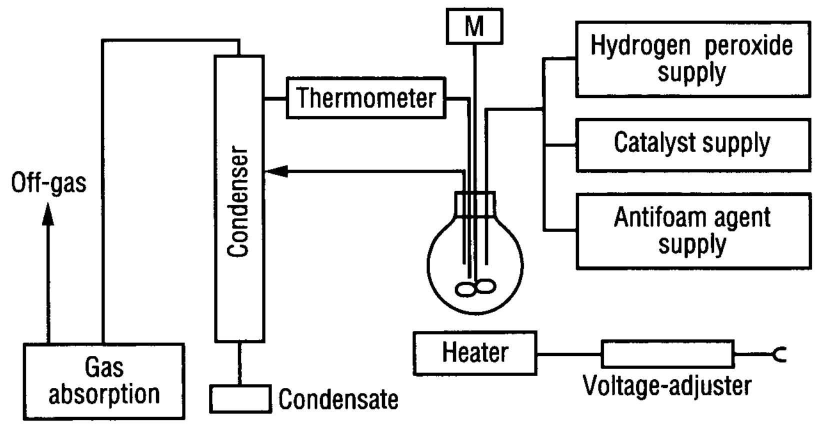

Fig. 2 Sketch of laboratory resin oxidation system.

whereas O_2, which was the direct decomposition product of hydrogen peroxide, increased gradually; the reaction ended with the formation of a reddish solution in about 3.5 h. A typical gas releasing process over time is given in Fig. 3. When a small amount of NaOH solution (5%) is added, the solid beads stuck on the surface of the flask were dissolved, and NH_3 was released simultaneously. If sufficient NaOH was added, the solution became basic and was no longer beneficial to the decomposition of resin because of the violent direct decomposition of hydrogen peroxide.

(2) Cation-exchange resin can be dissolved thoroughly when the oxidation process is catalyzed by Fe^{2+}, Ni^{2+}/Cu^{2+}, Mn^{2+}/Cu^{2+}, and Cu^{2+}. [Cu^{2+} was the most effective catalyst for anion resins. In addition, Cu^{2+} can also change cation resins from a solid to a liquid state; however, it cannot thoroughly decompose the cation resin.] The best catalyst was the Fe^{2+} system because of the higher efficiency of hydrogen peroxide and fast dissolution of resin when the reaction began; about 20 min later, all resins turned into a black solution. With the addition of hydrogen peroxide and catalyst, the color of the solution became lighter, and the final color was yellow. For the other systems, the residual liquid was almost colorless, but the chemical oxygen demand (COD) values were higher than those in the Fe^{2+} system. The system of Ni^{2+} and Mn^{2+} alone could not decompose the cation resin completely but could only change resins to a liquid state.

(3) For mixed resins, decomposition catalyzed by Fe^{2+}/Cu^{2+} was most efficient, and the results were similar to those of the anion-exchange resin with more solid residuals formed.

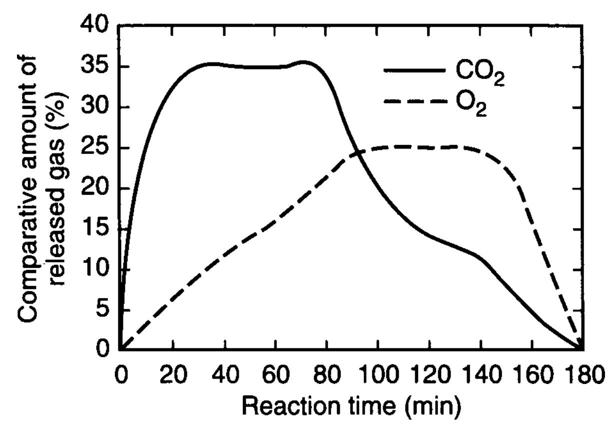

Fig. 3 CO_2 and O_2 release history in decomposition reaction of anion resin.

Factor Analysis

Temperature. The desired mixing temperature for the wet catalytic oxidation exchange process is generally from 97 to 99 °C, just less than the initial boiling point of water. Direct decomposition of hydrogen peroxide is comparatively violent when the temperature is lower, whereas excessively high temperatures cause such problems as foaming and diffusing of nuclides to the gas phase. Figure 4 shows the reactions between two temperature ranges.

Amount of Hydrogen Peroxide. As an oxidant, hydrogen peroxide plays a key role in the decomposition reaction. Adding more hydrogen peroxide results in more resins being completely decomposed, thereby increasing the cost of the treatment. It would be better

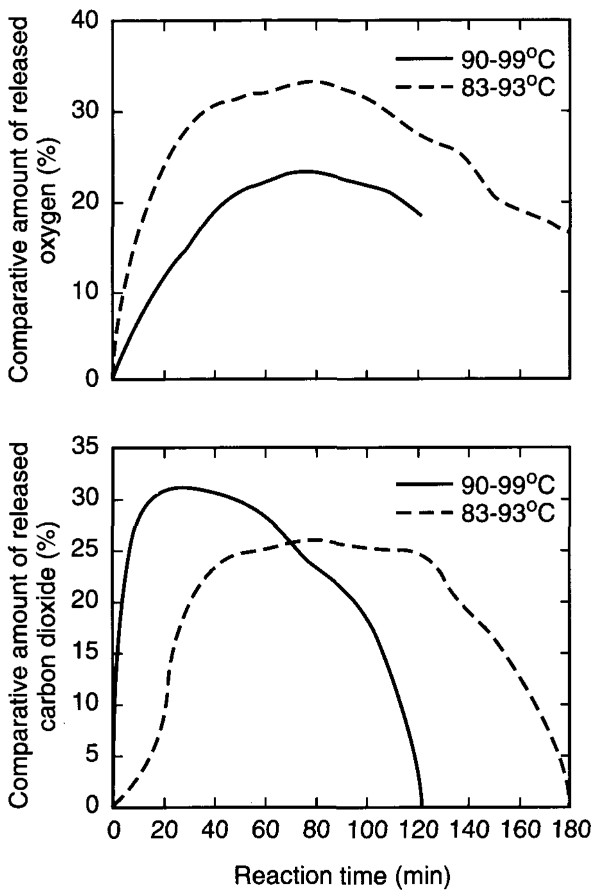

Fig. 4 **Effect of temperature on resin degradation.**

to add hydrogen peroxide continuously to raise its efficiency. Figure 5 shows the relationship between the dosage of hydrogen peroxide and the total organic carbon (TOC) value of the reaction residual liquid for anion resins.

Catalyst. Catalysis plays an important role in the decomposition reaction and obviously influences the course of the reaction. Table 1 indicates that Fe^{2+} alone is an effective catalyst in the degradation of cation-exchange resin, while Cu^{2+} is effective for anion resin, and Fe^{2+}/Cu^{2+} is preferable for mixed resins. During the reaction process, some oxidation products

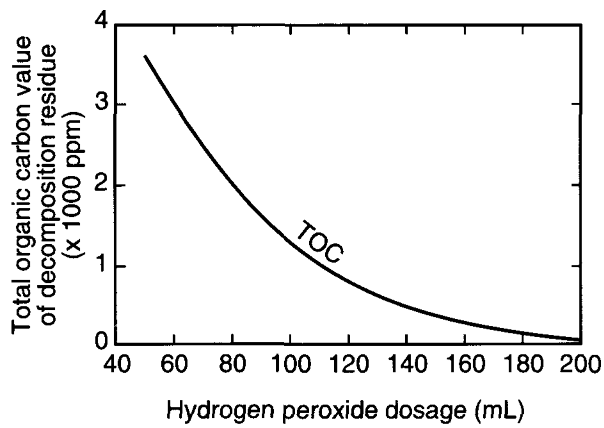

Fig. 5 Relationship between amount of dosing hydrogen peroxide and total organic carbon (TOC) value of reaction residual liquid for anion resin.

Table 1 Experimental Results of Different Catalytic Systems

Number	Weight of wet resin, *g*	Amount of catalyst,[a] *mmol*	Amount of hydrogen peroxide, *mol*	COD value of residual liquid, *mg/L*	Total COD of residual liquid,[b] *mg*	Weight of residual solid, *g*
1	15 (anion)	A: 1.50 C: 1.50	1.20	1 000 to 1 500	150 to 190	0.3 to 0.7
2	15 (anion)	B: 1.50 C: 1.50	1.20	1 000 to 1 500	150 to 190	0.3 to 0.7
3	15 (anion)	D: 1.50 C: 1.50	1.20	800 to 1 300	130 to 170	0.3 to 0.7
4	15 (anion)	C: 1.50	1.20	800 to 1 300	130 to 170	0.3 to 0.7
5	15 (cation)	A: 1.50	0.90	13 000 to 17 000	1 500 to 1 900	0
6	15 (cation)	B: 1.50	0.90	12 000 to 16 000	1 400 to 1 800	0
7	15 (cation)	C: 1.50	0.90	2 000 to 4 000	280 to 360	0.01 to 0.02
8	15 (cation)	D: 1.50	0.90	<100	12 to 20	0
9	25 (cation: anion = 2:1)	C: 2.50	1.70	2 000 to 4 000	300 to 700	0.5 to 1.0
10	25 (cation: anion = 2:1)	D: 2.50 C: 2.50	1.70	1 000 to 2 000	150 to 450	0 5 to 1.0

[a]A, $Ni(NO_3)_2$; B, $Mn(SO_4)$; C, $Cu(NO_3)_2$; D, $FeSO_4$.
[b]Total COD = Chemical oxygen demand (COD) value of the residual liquid times the volume of the residual liquid.

such as aromatic acids and organic amines could react with the catalyst and produce some complex compounds simultaneously. To avoid these reactions, which consumed catalyst and thereby hindered the hydroxyl radical chain reactions, the catalyst was fed continuously. The more the catalyst was fed, the more complete the decomposition of resins.

Agitation. Agitation has three functions: (1) mixing the reactants to form a homogeneous solution and improving the reaction rate, (2) preventing foaming, and (3) resisting producing sticky substances or crushing these sticky substances to enhance the degradation reaction continuously.

Soaking Resin by Catalyst. Experimental results showed that using parts of the catalyst to presoak the resin for some time (20 h was sufficient) could bring about a faster initial and more complete degradation reaction. The utility of soaking was more obvious for cation resin. The COD value of the residual liquid was less than 100 mg/L for soaked cation resins; it was more than 250 mg/L for nonsoaked ones.

NaOH Solution. During the decomposition of anion resin, adding some NaOH solution will reduce the amount of the solid residuals and raise the pH value of the system and thus simultaneously accelerate the direct decomposition of hydrogen peroxide and decrease its efficiency. According to the orthogonal tests, the addition of 0.7 g NaOH was suitable for 15 g wet anion resin, and the pH value of the reaction liquid was about 5.

Orthogonal Tests

Orthogonal tests investigated the influence on the experimental results of the dosage of hydrogen peroxide, catalyst, and NaOH. Here TCOD (total COD of the liquid residual = COD value × the volume of residual liquid) and TORG (= TCOD × weight of solid residual) were chosen as compressive assessment indexes (CAIA and CAIB). The less the index, the better the experimental result—for example, a smaller TCOD value indicates less solid residue. A factor analysis (a mathematical method to perform orthogonal tests) was performed to determine which factor influences the result the strongest. This showed that the amount of Cu^{2+} and hydrogen peroxide were the most important factors; this point was also confirmed by the experimental phenomenon. Increasing the amount of catalyst can decrease the amount of hydrogen peroxide required, which is expensive. In this study, the ratio of the amount of hydrogen peroxide to catalyst was kept as small as practical in order to save on treatment costs; however, if the concentration of the catalyst was too large, the reaction became so violent that it produced more solid residue and influenced the solidification.

The operating conditions and results of resin decomposition are listed in Table 2.

The radioactivity analysis results obtained during the spent resin decomposition process indicated that the radioactive nuclides loaded in the spent resins remained in the decomposition solution and solid residues—no radioactivity was detected in the off-gas.

Table 2 Operating Condition and Results of Resin Decomposition

Item	Cation (732)[a]	Anion (711)[a]	Mixed[b]
Pure H_2O_2/dried resin, kg/kg	3.5	4.5	3.7
Catalyst/resin reaction time, kg/kg·min	A: 1.67×10^{-4}	A: 0 B: 1.49×10^{-4}	A: 1.52×10^{-4} B: 1.52×10^{-4}
Antifoaming agent/resin, L/kg	0	0.01	0.01
Temperature, °C	97 to 99	97 to 99	97 to 99
Reaction time, h	2.5	3.5	2.5 to 3.0
COD value of liquid residual, mg/L	<300	<3000	3000 to 4000
pH value of liquid residual	1.0 to 1.5	4.0	2.0
Decomposition ratio,[c] %	~100	>90	>85
H_2O_2 efficiency, %	75 to 85	85 to 90	>80

[a]A, $FeSO_4$; B, $Cu(NO_3)_2$. The resins 732 and 711 are strong acidic and basic polystyrene resins, respectively.

[b]Weight ratio of cation:anion was 2:1.

[c]Decomposition ratio = weight of solid residue in dried state/weight of dissolved resin in dried state.

For the preparation of these decomposition residues for cementation, it was necessary to neutralize them to pH values of 8 to 10 by NaOH solution and then reduce the volume by evaporating it at 99 °C until the salt content of evaporated residue is up to 40 wt %, which is the highest salt content for cementation. These preparations must be made because of the limitation of salt content required for cementation. Three cement matrices in solidification were chosen for immobilization of decomposition residues. These were sulfate-resistant cement (SRC), acrylate copolymer (ACP–SRC), and epoxide plastic–polyamide–styrene (EPPAS–SRC). The parameters and product properties are listed in Table 3.

In this study, the amount of solid residuals varied from 0.2 to 0.7 g per 15 g wet anion resins, and the performance of cementation product was good even if it contained 0.7 g solid. The experimental results showed that if the total COD value of residual liquid was less than 0.6 g for 15 g wet anion resin and 0.06 g for 15 g wet cation resin, the cement solidification process was not affected and the ratio of TCOD to cement was less than 0.04 for anion resin.

The initial evaluation shows that the total treatment and disposal cost of direct cementation is about 170 000 ¥ (= $2 050 U.S.) per cubic meter of spent resin. By wet catalytic oxidation, however, treatment and disposal costs are only about 80 000 ¥ (= $960 U.S.) per cubic meter of spent resin. The result is comparable with that of Place.[2]

HYPOTHESIS ON THE MECHANISM OF WET CATALYTIC OXIDATION OF IER BY HYDROGEN PEROXIDE

Because all the metal ions used as catalysts in this study have different valences, it can be initially inferred that the mechanism of catalysis is as follows:

$$M^{n+} + H_2O_2 \rightarrow M^{(n+1)+} + OH^- + HO\bullet \tag{20}$$

$$HO\bullet + M^{n+} \rightarrow M^{(n+1)+} + OH^- \tag{21}$$

$$M^{(n+1)+} + H_2O_2 \rightarrow M^{n+} + HOO\bullet + H^+ \tag{22}$$

$$M^{(n+1)+} + HOO\bullet \rightarrow M^{n+} + O_2 + H^+ \tag{23}$$

$$R\bullet + Cu^{2+} \rightarrow Cu^+ + R^+ \tag{24}$$

$$Cu^+ + M^{(n+1)+} \rightarrow Cu^{2+} + M^{n+} \tag{25}$$

Table 3 Solidification Parameters and Product Properties

Cement polymer	Cement matrices: Sulfate-resistant cement (SRC)	Acrylate copolymer (ACP–SRC)	Epoxide plastic–polyamide–styrene (EPPAS–SRC)
Polymer content	0	4%	8%
Water/cement (in weight)	0.5	0.5	0.5
Salt/cement (in weight)	0.35	0.35	0.35
Bleed	No	No	No
Set time, h			
Initial	2.0	1.6	1.9
Final	2.2	2.1	2.3
Compressive strength (MPa)[a] maintained for 28 d	40.0	30.8	41.3
irradiated[b]	35.0	29.4	38.7
High-temperature stability[c]	Fine	Fine	Fine
Accumulative leaching ratio of total β for 42 d,[d] cm	9.98×10^{-2}	7.45×10^{-2}	7.17×10^{-2}
Density, $g \cdot cm^{-3}$	2.06	2.03	1.92
Volume reduction factor (VRF)[e]	0.39	0.37	0.34

[a]Tested according to National Standard GB-177-62 "Physical Test for Cement."
[b]Total irradiated γ dosage: 2.8×10^5 Gy.
[c]Tested according to ASTM D63-74.
[d]Tested according to National Standard 7023-86.
[e]VRF = (original volume of dissolved resin – volume of final cemented product)/original volume of dissolved resin.

(In these equations, n equals 2 for iron, nickel, and manganese; whereas for copper, n equals 1 except in Eq. 25.)

Because the oxidation potential between Mn^{2+} and Mn^{3+} is the largest among these ions, if the total amount of catalyst added to the system is the same, the concentration of Mn^{2+} will be the largest, whereas Fe^{2+} will be the smallest for two-valence ions. Fe^{3+} will be the largest for three-valence ions, whereas the concentration of Mn^{3+} and Ni^{3+} will be very small (they are very unstable in aqueous solution). The ions can effectively catalyze the reaction until they are absorbed onto the exchange site. For cation resins, the higher the valence of the ion, the more easily they can be absorbed. Because the concentration of Fe^{3+} is the highest, the reaction is fastest, the decomposition is the most complete, and the medium products are easy to oxidize continuously. For Mn^{2+} and Ni^{2+} systems, some medium products, such as acetone and acetic acid, hindered the chain reactions. Adding Cu^{2+} oxidizes these substances and produces some sediment complex compounds of copper.

Equation 20 benefits the decomposition process, and Eq. 21 hinders that process because the former produces the hydroxyl radical and the latter consumes it. One of the practical methods is to continuously add hydrogen peroxide.

Because the $-CH_2-$'s p orbital can conjugate with benzene's π orbital and becomes a delocalized π-bond, thereby dispersing electrons, the benzyl of anion resin is a stable free radical. Conversely, the N^+ attracts electrons, whereas the $-CH_2-$ group releases electrons, which causes the electron cloud, including the delocalized π-bond, to move toward N^+. The quaternary-amine group separates from the benzene at first and then is oxidized by the hydroxyl radical, releasing some amine, which benefits foam producing and forms NH_4^+. Some NH_4^+ copolymerizes to solid residue and releases NH_3 when reacting with NaOH. The rest reacts with organic acid, which is the oxidation product of the resin and is soluble and thus emits NH_3 when NaOH is added to the liquid residue.

Infrared and ultraviolet spectra of anion and cation resins and their decomposition residues show that there are more than ten kinds of aromatic compounds in the reaction liquid and solid residue of anion resin, whereas none were formed from the cation resin except simple organic ones. This can also be proved by the COD values of liquid residues. It is more difficult to oxidize anion and mixed resins than cation resins. The reasons are as follows: (1) The S atom releases electrons, whereas the hydroxyl radicals want electrons; thus it makes the S atom and hydroxyl radicals approach and react easily. The SO_3H group separates from benzene and becomes sulfuric acid, and the benzene goes on decomposing, which results in no aromatic compounds remaining in the liquid residue for cation resins. The N^+ attracts electrons, and the benzyl radical of the anion is stable, which makes it difficult to react with the hydroxyl radical. (2) These compounds produced in the decomposition of anion resin are strong, complex agents for metal ions, which can impair the efficiency of the catalysts. (3) Quaternary-amine substances are oxidized slowly and will decompose with difficulty. (4) Copolymerization accompanies the decomposition of resins.[5] These sticky substances and floating solids produced at the later stage of reaction are relevant to points 3 and 4.

CONCLUSIONS

1. Radioactive spent ion-exchange resin can be successfully treated and dissolved by H_2O_2–Fe^{2+}/Cu^{2+}, H_2O_2–Mn^{2+}/Cu^{2+}, H_2O_2–Ni^{2+}/Cu^{2+}, and H_2O_2–Cu^{2+} systems. Fe^{2+} is the most effective catalyst for the decomposition of cation resin, and Cu^{2+} is the most effective catalyst for anion resin. The best decomposition results are obtained from the Fe^{2+}/Cu^{2+} system for mixed resins. The resins transform from a solid phase consisting of an organic matrix into a liquid phase containing a small amount of organic components, and the decomposition ratio is approximately 100% for cation resin, more than 90% for anion resin, and more than 85% for mixed resin.

2. The radioactive nuclides loaded in the spent resin during the period of decomposition are concentrated completely in the decomposition solution and solid residue. No radioactive contamination is associated with the off-gas, so it can be vented directly to the atmosphere without any further treatment.

3. The concentrated decomposition residue can be successfully immobilized in cement, and the cemented products in terms of quality meet regulatory requirements stipulated by the International Atomic Energy Agency for long-term storage. The total disposal cost will run about 50% less than that for direct solidification.

4. The volume reduction percentage of the H_2O_2 oxidation process is up to 30 to 40% compared with

the volume of directly cemented ion-exchange resin, which has a volume increment of 80% (Ref. 6).

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A Comparison Study and Resolution of Differences Between Emergency Response and Safety Analysis Codes Used at the Savannah River Site

By A. A. Simpkins[a]

Abstract: *The Savannah River Site uses different dose assessment codes for safety analysis and emergency response. Both models contain a Gaussian plume dispersion model, but there are several inherent differences between the codes. Comparisons using the same input show that the two codes produce doses that differ by less than 3%; however, conditions exist in which the codes give significantly different results.*

Savannah River Site (SRS) has many characteristics that make it unique in such areas as safety analysis and emergency preparedness. One such characteristic is the large area of the site, almost 300 square miles, which greatly reduces the potential effects on off-site individuals from atmospheric releases that could occur near the center of the site. Also, the site has been in operation for more than 40 years, and a wealth of site-specific data is available for use in the dose assessment methodologies.

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Different codes have been developed at the SRS to address real and hypothetical accidents. For incidents that involve releases to the atmosphere, PUFF–PLUME (Ref. 1) was developed to make decisions regarding evacuation or sheltering of on-site and off-site individuals by accessing real-time meteorology from seven different on-site meteorological towers. PUFF–PLUME allows a choice between a Gaussian plume model and a Gaussian puff model. Both wet and dry depositions may be considered. Only the inhalation exposure pathway is used.

Currently, during the preparation of a Hazards Assessment Document (HAD), PUFF–PLUME is used to determine the dose to the maximally exposed individual. An Emergency Management Guide[2] (EMG), which provides guidance for complying with Department of Energy (DOE) Order 5500.3A, states that "...consequence assessment models used for emergency planning and response purposes at the facility should be used to conduct this hazards assessment." Another section of the EMG specifies the use of dose that is not exceeded 95% of the time on the basis of historical meteorological conditions (when a 5-year joint frequency distribution of meteorological conditions is used, the dose reported is expected to be exceeded only 5% of the time); however, PUFF–PLUME was designed to access real-time meteorology or use a specific stability class and wind speed combination. Therefore the code cannot determine a dose that is exceeded 95% of the time on the basis of a historical meteorological frequency distribution. Because the guidance requires the use of PUFF–PLUME and the determination of doses that are not exceeded 95% of the time on the basis of historical meteorological conditions for hazards assessment, a specific stability class and wind speed combination was originally assigned to represent these conditions.

The computer code AXAIR89Q (Refs. 3 and 4) is primarily used to produce documentation for safety analysis and predictive purposes and accesses a 5-year, historical meteorological database. AXAIR89Q strictly follows the guidance in U.S. Nuclear Regulatory Commission (USNRC) Regulatory Guide 1.145 (Ref. 5). AXAIR89Q contains a Gaussian plume model with both inhalation and plume shine exposure pathways. Deposition is not incorporated into the code. The code can calculate doses that are not exceeded either 99.5% or 50% of the time on the basis of historical meteorological conditions.

When AXAIR89Q was compared with PUFF–PLUME, doses that were not exceeded 95% of the time on the basis of historical meteorological conditions calculated by PUFF–PLUME were sometimes higher than the doses that were not exceeded 99.5% of the time calculated by AXAIR89Q. A study was initiated to determine the differences between the two codes.

MODEL DIFFERENCES

The computer codes AXAIR89Q and PUFF–PLUME differ in many areas because of different equations that are used to determine various parameters within each of the two codes. Each of the differences is discussed in detail in the following text.

Diffusion Coefficient Relationships

PUFF–PLUME and AXAIR89Q apply different diffusion coefficients. PUFF–PLUME uses Pasquill–Briggs coefficients,[6,7] whereas AXAIR89Q uses Pasquill-Gifford coefficients[8] as depicted in the Turner Workbook[9] curves. The use of different diffusion coefficients can result in considerable differences, depending on the stability class and wind speed combination. Figure 1 shows the ratio of relative air concentrations (χ/Q), using Pasquill–Briggs diffusion coefficients vs. Pasquill–Gifford diffusion coefficients for a release height of 10 m (Ref. 10). The use of Pasquill–Briggs

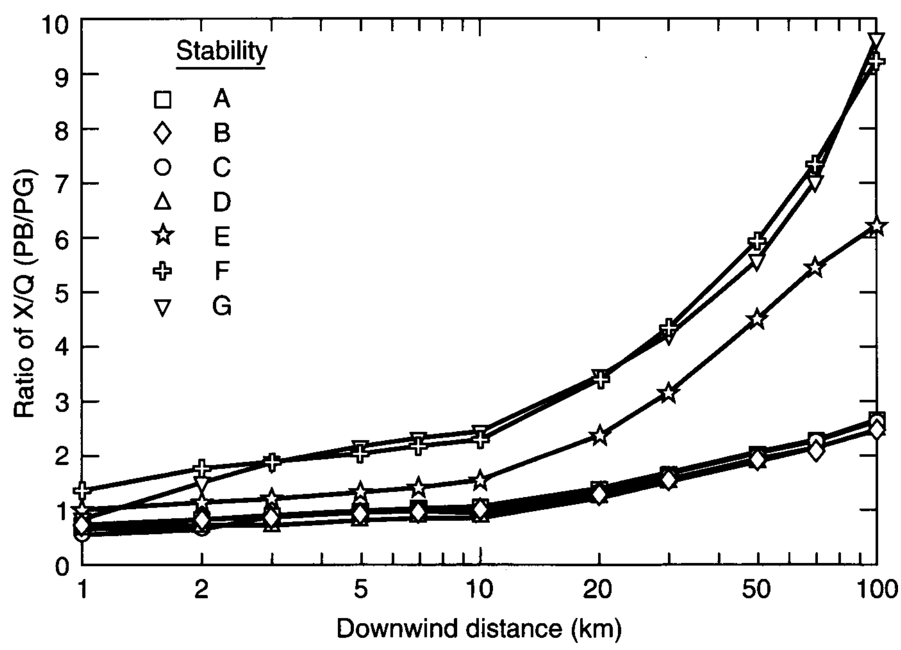

Fig. 1 Ratio of relative air concentrations (χ/Q) using Pasquill–Briggs (PB) vs. Pasquill–Gifford (PG) diffusion coefficients (σ_z unlimited). The atmospheric stability classes are the Pasquill categories defined in USNRC Regulatory Guide 1.23 (Ref. 11).

diffusion coefficients can cause as much as a factor of 2 increase in relative air concentration at 10 km when compared with Pasquill–Gifford diffusion coefficients.

Correction of Release Height for Terrain Effects

PUFF–PLUME does not consider terrain effects. AXAIR89Q takes into account the terrain height in determining the effective height of the release using Eq. 1.

$$H_e = H_s - H_t \qquad (H_t < H_s)$$
$$H_e = 0 \qquad (H_t \geq H_s) \tag{1}$$

where H_e is the effective release height, H_s is the stack release height, and H_t is the terrain height for the given receptor location.

The terrain height at the receptor location is defined as the maximum height difference between the receptor and the release location.[5] In AXAIR89Q, depending on which sector is selected, the terrain can vary by as much as 40 m at 15 km (approximate site boundary distance) from the source. As the vertical diffusion coefficient decreases and the terrain height increases, the differences become more significant.

Treatment of Fumigation Conditions

On a clear morning shortly after the sun rises, the inversion present just above the top of the stack acts as a lid to the shallow unstable layer next to the ground. This condition is known as fumigation. Fumigation occurs in stable conditions, and vertical spreading is more prominent on the lower side of the plume rather than on the upper. PUFF–PLUME does not implicitly include fumigation, but the user is expected to choose the appropriate stability class and wind speed class to analyze these effects. AXAIR89Q follows the guidance of USNRC Regulatory Guide 1.145, which states that, for inland sites such as SRS, fumigation is allowed to occur 25% of the time for the 2-h release period.[5] For SRS, the following conditions must be met for fumigation to occur:

1. Atmospheric conditions are stable (stability categories E, F, or G).
2. Wind speed at the release height is less than 4 m/s.
3. "...[E]quation 2 (see below) should be used instead of equation 3 (see below) at distances greater than the distance at which the χ/Q values determined using equation 2 with $H_e = 0$ and equation 3 are equal" (Ref. 5).

Only when all three of these conditions are met will the fumigation algorithm be invoked. Equation 2 is used for nonfumigation conditions.[5]

$$\frac{\chi}{Q} = \frac{e^{(-h_e^2/2\sigma_z^2)}}{\pi\sigma_y\sigma_z U_h} \tag{2}$$

where χ/Q = relative air concentration—ratio of concentration of released material in air to the release rate of the material, s/m^3
h_e = effective stack height (stack height–terrain height), m
σ_y = lateral plume spread, m
σ_z = vertical plume spread, m
U_h = wind speed representing conditions at the release height, m/s

Equation 3 is used for fumigation (Ref. 5).

$$\left(\frac{\chi}{Q}\right)_f = \frac{1}{(2\pi)^{1/2}\sigma_y h_e U_{h_e}} \tag{3}$$

where U_{h_e} is the wind speed representative of the fumigation layer of depth h_e (m/s). The fumigation χ/Q is used only when it exceeds the nonfumigation χ/Q.

Interpolation of Results

Because PUFF–PLUME dose calculations are performed on the basis of a specific stability class and wind speed combination, no interpolation is needed. AXAIR89Q determines the dose not exceeded 99.5% of the time on the basis of historical meteorological conditions for each of the 16 sectors and selects the highest as the dose for the given distance. In each sector the 42 doses determined by the 7 stability classes (A–G) and 6 wind speed classes are ranked from highest to lowest along with their frequency of occurrence. Wind speed classes 1–6 correspond to the following ranges: 0–2, 2–4, 4–6, 6–8, 8–12, and >12 m/s. A cumulative frequency is associated with each of the 42 doses. When the cumulative frequency exceeds 0.5%, interpolation of dose is performed. The wind speed and stability class combinations used for the interpolation are not likely to have the same stability class or wind speed. The interpolation will contribute some error.

Table 1 is a sample of the ranking of dose along with corresponding cumulative frequency, stability class, and wind speed for a particular sector. For the case shown, the interpolation occurs between the two marked cases (*), which correspond to stability class F with wind speed class 4 and stability class E with wind speed class 2. Interpolating between these two doses to determine the 0.5% cumulative frequency results in a dose of 958 mrem.

The doses cannot be determined correctly by choosing an intermediate class; for example, an *incorrect* comparison would be to choose an intermediate class for comparison of either stability E or F with a wind speed class of 3. Notice that one of these two combinations of stability and wind speed classes results in a dose with a cumulative frequency of less than 0.5%. The other combination corresponds to a dose representative of 97.75% meteorological conditions. For these reasons, an intermediate class cannot be chosen to be representative of doses not exceeded 99.5% of the time on the basis of historical meteorological conditions.

Table 1 Sample Ranking of AXAIR89Q Dose

Dose, mrem	Cumulative frequency, %	Stability class	Wind speed category
$8.93 \times 10^{+3}$	0.007	G	1
$3.65 \times 10^{+3}$	0.014	G	2
$3.06 \times 10^{+3}$	0.021	F	1
$2.78 \times 10^{+3}$	0.041	G	3
$1.67 \times 10^{+3}$	0.148	F	2
$1.20 \times 10^{+3}$	0.370	F	3
$1.10 \times 10^{+3}$	0.386	E	1
$*1.01 \times 10^{+3}$	0.404	F	4*
$*6.27 \times 10^{+2}$	1.077	E	2*
$4.45 \times 10^{+2}$	2.261	E	3
$4.24 \times 10^{+2}$	2.316	D	1
$3.54 \times 10^{+2}$	2.359	F	4
$2.96 \times 10^{+2}$	2.428	C	1
$2.29 \times 10^{+2}$	2.505	B	1
$2.24 \times 10^{+2}$	3.430	D	2
$1.84 \times 10^{+2}$	3.788	A	1
$1.52 \times 10^{+2}$	5.239	D	3

Initial Source Size

In PUFF–PLUME, the user is allowed to enter the initial plume size. The initial dimensions of the plume are input as σ_{oy} by σ_{oz} (in meters), and the value of σ_y is determined by Eq. 4.

$$\sigma_y = \left(\sigma_{yPB}^2 + \sigma_{oy}^2\right)^{1/2} \tag{4}$$

where σ_{yPB} value is determined by using the Pasquill–Briggs equations.

The value for σ_z is determined in the same manner. The default values of initial plume size in PUFF–PLUME are 3 m (σ_{oy}) by 3 m (σ_{oz}). In AXAIR89Q, the initial source size is assumed to be infinitesimally small. For relatively unstable categories with large values of σ_y and σ_z, the initial plume size becomes negligible. In the classes that are more stable where σ_z can be as low as 10 m, the initial plume size affects the results.

Treatment of Inversion Height

In PUFF–PLUME, the user has the option of entering the inversion height (H_{inv}). In AXAIR89Q, the inversion or lid height is set to a constant value of 200 m. In both codes, the value of the vertical diffusion coefficient (σ_z) is allowed to be no greater than the product of 0.8 H_{inv}. Therefore, even though H_{inv} is not directly used to determine the relative air concentration, it can have an impact on the resulting doses. The limitation of vertical diffusion coefficients would have the greatest effect for the unstable classes (A and B) and possibly at greater distances (d > 3 km) for the intermediate classes (C and D). Stability classes E and F should not be affected.

Consideration of Inhalation and Shine Doses

PUFF–PLUME considers only inhalation dose, whereas AXAIR89Q considers both inhalation and plume shine doses. Depending on the isotopes considered, this can have an effect on the differences in doses produced by the two models.

MODEL COMPARISONS

The two codes were compared with three degrees of rigor to determine the differences. In Case 1, a standard AXAIR89Q calculation was compared with a wind speed and stability combination within PUFF–PLUME that was thought to correspond to a dose not exceeded 95% of the time on the basis of historical meteorological conditions. In Case 2, the dose for 99.5% meteorological conditions from AXAIR89Q was compared with the PUFF–PLUME dose resulting from the same stability and wind speed combinations as used for interpolation within AXAIR89Q. Finally, in Case 3, internal modifications were made to AXAIR89Q and to certain data files to allow an extremely rigorous comparison. For each of the comparisons, the deposition model in PUFF–PLUME was not invoked.

Case 1

The dose not exceeded 95% of the time on the basis of historical meteorological conditions was needed for hazard assessment documentation. Because PUFF–PLUME does not have this capability, a stability and wind speed combination "typical" of 95% meteorology was assigned. This selection was F stability class and 1 m/s wind speed. Although this may be representative of 95th percentile doses, there is *not* a unique combination that will always result in the dose not exceeded 95% of the time on the basis of historical meteorological conditions. AXAIR89Q was executed to determine the dose for meteorological conditions not exceeded 99.5% of the time. Table 2 shows the input values used for the comparison.

The results of the comparison of effective dose equivalents (EDEs) at the site boundary (11.9 km from release point) are shown in Table 3. The differences are remarkable; the results from PUFF–PLUME are actually higher than those from AXAIR89Q because stability class F at 1 m/s does not correspond to 95% meteorology for this particular case. Table 1 shows the actual values and stability class and wind speed combinations used for the interpolation. Stability class F with a wind speed of 1 m/s (wind speed category 1) actually corresponds to a cumulative frequency of 99.98%. These results demonstrate that no specific stability and wind speed combination can be selected before running AXAIR89Q for the comparison.

Case 2

Case 2 uses Table 1 data to select 99.5% probability. As part of the output, AXAIR89Q shows the two sets of stability class and wind speed combinations whose corresponding doses were interpolated between stabilities to determine the dose not exceeded 99.5% of the time on the basis of historical meteorological conditions. The dose at the site boundary was determined by interpolating between F stability with 6 to 8 m/s wind speed and E stability with wind speed between 2 and 4 m/s (see Table 1). These doses correspond to conditions not exceeded 99.6 and 98.92% of the time, respectively. This bracketing combination was used for the PUFF–PLUME comparison. These resulting doses at the site boundary are depicted in Table 4.

Table 2 Input Parameters for AXAIR89Q and PUFF–PLUME Comparison

Parameter	AXAIR89Q	PUFF–PLUME
Release area	H	H
Vent height, m	0	0
Dose factor library	ICRP 30	ICRP 30
Isotope	^{238}Pu	^{238}Pu
Amount of release, Ci	1.00	1.00 entered as 1.390×10^{-4} Ci/s
Duration of release, min	120	120
Inversion height, m	200	200

Table 3 Case 1 EDE[a] Comparison from AXAIR89Q and PUFF–PLUME

Distance	EDE, rem: AXAIR89Q	EDE, rem: PUFF–PLUME	Percent difference (AX-PF)/AX
100 m	641	8170	–1175
Site boundary	0.958	7.5	–683

[a]Effective dose equivalent.

Table 4 Comparison of PUFF–PLUME and AXAIR89Q at the Site Boundary

Stability class at wind speed	EDE[a], rem: AXAIR89Q	EDE[a], rem: PUFF–PLUME	Percent difference
F at 7 m/s	1.01	1.08	–7
E at 3 m/s	0.627	0.670	–7

[a]Effective dose equivalent.

Small differences in dose results still exist. One contributing factor was the wind speed values used in each of the codes. AXAIR89Q applies a historical average wind speed (based on actual data), whereas the Case 2 comparison applies the midpoint of the range for input into PUFF–PLUME. Another contributing factor is different diffusion coefficients applied within each of the codes; however, this case shows that AXAIR89Q and PUFF–PLUME were in much closer agreement.

Case 3

Case 3 compared the two models under identical conditions. The parameters used for the comparison were carefully chosen so that differences in input between the two models could be minimized. Internal modifications also were necessary to conduct a true comparison.

The most obvious difference, the meteorology, was examined first. Input to PUFF–PLUME is in the form of specific values of the standard deviation of the horizontal (σ_θ or σ_a) and vertical (σ_e) wind direction and the wind speed. In contrast to PUFF–PLUME, AXAIR89Q does not have the option of entering a specific stability class and wind speed combination. Instead, dose calculations are automatically made for a range of stability and wind classes and an exceedance probability (either 99.5% or 50%) is selected. Thus, in general, a rigorous comparison is not possible.

For a comparison with PUFF–PLUME, the meteorological joint frequency distribution accessed by AXAIR89Q was modified to set the frequency to zero for the entire distribution except for the category with the desired wind speed, direction, and stability class, which is set to 0.5%. This is the category the model will then select as the 99.5% meteorological conditions for the selected sector. AXAIR89Q prints the frequencies as a function of wind speed, direction, and stability class as part of the output, so the correct category was easily verified. Data concerning the wind speeds corresponding to the specific category were also changed to agree with the input wind speed to PUFF–PLUME.

Another minor adjustment made in the AXAIR89Q input is to select a stack release height equal to that used by PUFF–PLUME. In AXAIR89Q, an adjustment is automatically made to the stack release height to account for the terrain, as per USNRC Regulatory Guide 1.145 (Ref. 5). The effective stack height is the release height minus the maximum terrain height between the release point and the receptor. For the Case 2 comparison, the difference was 2.74 m. For Case 3, the stack release height in AXAIR89Q was increased 3 m (AXAIR89Q will accept only whole numbers for release height), so the effective stack height is similar to the value used in the PUFF-PLUME case.

Diffusion coefficients in the current operational version of AXAIR89Q are different from those in PUFF–PLUME. A test version of AXAIR89Q was created with the diffusion coefficients changed to Pasquill[7] for σ_y and Briggs[6] for σ_z. PUFF–PLUME assumes six stability classes (A–F), and AXAIR89Q assumes seven (A–G). AXAIR89Q is modified to use the same equations for the diffusion coefficients for stability classes F and G. This did not add much error because both are stable categories.

Tables 5 to 7 show the comparisons among doses determined by AXAIR89Q and PUFF–PLUME for the following stability class and wind speed combinations: class A at 6 m/s, class C at 4 m/s, and class E at 8 m/s. One curie of ^{131}I was used for the release amount for each of the Case 3 comparisons with the release height of 65 m for AXAIR89Q and 62 m for PUFF–PLUME. All other inputs are those listed in Table 2. These combinations were arbitrarily chosen from stability and wind speed combinations for which the fumigation algorithm is not invoked. All doses shown are 50-year committed EDEs.

Table 5 Dose Comparison of AXAIR89Q vs. PUFF–PLUME (Stability Class A, Wind Speed 6 m/s)

Distance, km	Dose, mrem AXAIR89Q	Dose, mrem PUFF–PLUME	Percent difference
1.8	9.96×10^{-3}	9.78×10^{-3}	1.8
7.6	3.35×10^{-3}	3.28×10^{-3}	2.1
10.8	2.56×10^{-3}	2.51×10^{-3}	2.0
14.1	2.25×10^{-3}	2.20×10^{-3}	2.2

Table 6 Dose Comparison of AXAIR89Q vs. PUFF–PLUME (Stability Class C, Wind Speed 4 m/s)

Distance, km	Dose, mrem AXAIR89Q	Dose, mrem PUFF–PLUME	Percent difference
1.4	4.10×10^{-2}	4.07×10^{-2}	0.7
7.2	8.55×10^{-3}	8.51×10^{-3}	0.5
9.4	6.96×10^{-3}	6.94×10^{-3}	0.3
14.4	5.45×10^{-3}	5.43×10^{-3}	0.4

Table 7 Dose Comparison of AXAIR89Q vs. PUFF–PLUME (Stability Class E, Wind Speed 8 m/s)

Distance, km	Dose, mrem AXAIR89Q	Dose, mrem PUFF–PLUME	Percent difference
1.4	3.11×10^{-2}	3.19×10^{-2}	–2.6
7.6	2.15×10^{-2}	2.16×10^{-2}	–0.5
10.1	1.60×10^{-2}	1.61×10^{-2}	–0.6
14.4	1.30×10^{-2}	1.30×10^{-2}	0.0

Although the models were in close agreement, the differences can still be attributed to minor differences within each of the models (i.e., terrain height and unit conversions).

CONCLUSIONS

The methodologies in AXAIR89Q and PUFF–PLUME are similar when consistent meteorological conditions are applied to both codes; however, in most cases the two models should not be compared directly. The differences result from the different functions of the two models and the invocation of special algorithms in AXAIR89Q.

In the future, no specific wind speed and stability class combination should be chosen for a comparison unless AXAIR89Q is executed first for the given set of input parameters and then PUFF–PLUME is executed with the correct combination. AXAIR89Q could be modified for use with HADs.

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Operating Experiences

Edited by G. A. Murphy

Reactor Shutdown Experience

Compiled by J. W. Cletcher[a]

This section presents a regular report of summary statistics relating to recent reactor shutdown experience. The information includes both numbers of events and rates of occurrence. It was compiled from data about operating events entered into the SCSS data system by the Nuclear Operations Analysis Center at the Oak Ridge National Laboratory and covers the six-month period of July 1–December 31, 1995. Cumulative information, starting from January 1, 1984, is also shown. Updates on shutdown events included in earlier reports are excluded.

Table 1 lists information on shutdowns as a function of reactor power at the time of the shutdown for both boiling-water reactors (BWRs) and pressurized-water reactors (PWRs). Only reactors in commercial operation at the start of the reporting period (July 1, 1995) are included. The second column for each reactor type shows the annualized shutdown rate for the reporting period. The third and fourth columns list cumulative data (numbers and rates) starting as of January 1, 1984.

Table 1 Reactor Shutdowns by Reactor Type and Percent Power at Shutdown[a] (Period Covered is the Second Half of 1995)

	BWRs (37)				PWRs (75)			
Reactor power (P), %	Number	Shutdown rate (annualized for period)	Cumulative number	Cumulative shutdown rate per reactor year[b]	Number	Shutdown rate (annualized for period)	Cumulative number	Cumulative shutdown rate per reactor year[c]
0	7	0.38	706	1.71	15	0.40	490	0.60
$0 < P \leq 10$	2	0.11	140	0.34	2	0.05	175	0.21
$10 < P \leq 40$	7	0.38	173	0.42	4	0.11	330	0.40
$40 < P \leq 70$	4	0.21	161	0.39	7	0.19	187	0.23
$70 < P \leq 99$	4	0.21	390	0.94	4	0.11	523	0.64
$99 < P \leq 100$	22	1.18	522	1.26	37	0.98	1247	1.52
Total	46	2.47	2092	5.06	69	1.83	2952	3.59

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

[b]Based on cumulative BWR operating experience of 413.40 reactor years.

[c]Based on cumulative PWR operating experience of 822.35 reactor years.

[a]Oak Ridge National Laboratory.

Table 2 shows data on shutdowns by shutdown type: *Shutdowns required by Technical Specifications* are automatic scrams under circumstances where such a shutdown was required; *Intentional or required manual reactor protection system actuations* are manual shutdowns in which the operators, for reasons that appeared valid to them, took manual actions to actuate features of the reactor protection system; *Required automatic reactor protection system actuations* are actuations that the human operators did not initiate but were required; *Unintentional or unrequired manual reactor protection system actuations* are essentially operator errors in which the human operators took action not really called for; and *Unintentional or unrequired automatic reactor protection system actuations* are instrumentation and control failures in which uncalled-for protective actuations occurred. Only reactors in commercial operation are included. The second column for each type of reactor shows the annualized rate of shutdowns for the reporting period. Cumulative information is shown in the third and fourth columns for each reactor type.

Table 3 lists information about shutdowns by reactor age category, both total numbers and rates in that category; it also shows cumulative results. Note that the age groups are not cohorts; rather reactors move into and out of the specified age groups as they age. The reactor age as used in this table is the number of full years between the start of commercial operation and the beginning of the reporting period (January 1, 1995, for this issue). The first line of this table gives the information for reactors licensed for full power but not yet in commercial operation on that date.

Table 2 Reactor Shutdowns by Reactor Type and Shutdown Type[a] (Period Covered is the Second Half of 1995)

	BWRs (37)				PWRs (75)			
Shutdown (SD) type	Number	Shutdown rate (annualized for period)	Cumulative number	Cumulative shutdown rate per reactor year[b]	Number	Shutdown rate (annualized for period)	Cumulative number	Cumulative shutdown rate per reactor year[c]
SDs required by Technical Specifications	9	0.48	277	0.67	12	0.32	426	0.52
Intentional or required manual reactor protection system actuations	7	0.38	215	0.52	22	0.58	417	0.51
Required automatic reactor protection system actuations	26	1.39	974	2.36	31	0.82	1653	2.01
Unintentional or unrequired manual reactor protection system actuations	0	0.00	9	0.02	2	0.05	22	0.03
Unintentional or unrequired automatic reactor protection system actuations	4	0.21	617	1.49	2	0.05	434	0.53
Total	46	2.47	2092	5.06	69	1.83	2952	3.59

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

[b]Based on cumulative BWR operating experience of 413.40 reactor years.

[c]Based on cumulative PWR operating experience of 822.35 reactor years.

Table 3 Reactor Shutdowns by Reactor Type and Reactor Age[a] (Period Covered is the Second Half of 1995)

Years in commercial operation (C.O.)	BWRs (37) Exposure during the period (in reactor years)	BWRs Number: Reactors	BWRs Number: Shutdowns	BWRs Shutdown rate (annualized for the period)	BWRs Cumulative number	BWRs Cumulative shutdown rate per reactor year	PWRs (75) Exposure during the period (in reactor years)	PWRs Number: Reactors	PWRs Number: Shutdowns	PWRs Shutdown rate (annualized for the period)	PWRs Cumulative number	PWRs Cumulative shutdown rate per reactor year
Not in C.O.[b]	0.500	1	0	0.00	330	20.65	0.000	0	0	0.00	336	34.24
First year of C.O.	0.000	0	0	0.00	121	9.00	0.000	0	0	0.00	281	9.96
Second through fourth year of C.O.	0.000	0	0	0.00	264	6.29	0.500	1	1	1.99	530	5.53
Fifth through seventh year of C.O.	0.500	1	2	3.97	187	4.25	2.540	7	3	1.18	335	3.12
Eighth through tenth year of C.O.	3.530	7	10	2.84	233	4.79	6.640	16	19	2.86	411	3.45
Eleventh through thirteenth year of C.O.	3.020	6	11	3.64	293	5.44	4.920	11	6	1.22	514	3.99
Fourteenth through sixteenth year of C.O.	0.180	1	1	5.53	401	6.16	3.530	7	10	2.84	387	3.20
Seventeenth through nineteenth year of C.O.	1.330	3	5	3.76	287	4.45	4.420	10	8	1.81	282	2.55
Twentieth through twenty-second year of C.O.	4.530	9	7	1.54	183	3.81	8.860	21	15	1.69	135	1.86
Twenty-third through twenty-fifth year of C.O.	3.530	7	8	2.27	71	3.04	4.830	12	5	1.04	49	1.95
Twenty-sixth through twenty-eighth year of C.O.	1.510	3	1	0.66	10	1.75	1.540	4	2	1.30	19	1.99
Twenty-ninth through thirty-first year of C.O.	0.000	0	0	0.00	9	3.00	0.000	0	0	0.00	5	1.67
Thirty-second through ninety-ninth year of C.O.	0.500	1	1	1.99	6	3.41	0.500	1	0	0.00	0	0.00
Total	19.140		46	2.40	2395	5.58	38.290		69	1.80	3284	3.93

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

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

Recent Developments

Edited by M. D. Muhlheim

Reports, Standards, and Safety Guides

By D. S. Queener

This article contains four lists of various documents relevant to nuclear safety as compiled by the editor. These lists are: (1) reactor operations-related reports of U.S. origin, (2) other books and reports, (3) regulatory guides, and (4) nuclear standards. Each list contains the documents in its category which were published (or became available) during the October 1995 through March 1996 reporting period. The availability and cost of the documents are noted in most instances.

OPERATIONS REPORTS

This category is listed separately because of the increasing interest in the safety implications of information obtainable from both normal and off-normal operating experience with licensed power reactors. The reports fall into several categories shown, with information about the availability of the reports given where possible. The NRC reports are available from the Nuclear Regulatory Commission (NRC) Public Document Room (PDR), 2120 L Street, NW, Washington, DC 20555.

NRC Office of Nuclear Reactor Regulation

The NRC Office of Nuclear Reactor Regulation (NRR) issues reports regarding operating experience at licensed reactors. These reports, previously published by the NRC Office of Inspection and Enforcement (IE), fall into two categories of urgency: (1) NRC Bulletins and Generic Letters, which require remedial actions and/or responses from affected licensees, and (2) NRC Information Notices and Administrative Letters, which are for general information and do not require any response from the licensee. The Administrative Letters, which contain information of an administrative or informational nature, were previously distributed under the generic letter category. No specific action is required in response to these Administrative Letters. The Generic Letters, Bulletins, and Information Notices are included in this issue.

NRC Generic Letters

NRC GL 95-08 10 CFR 50.54(p) *Process for Changes to Security Plans Without Prior NRC Approval,* October 31, 1995, 3 pages plus 25 pages of attachments.

NRC GL 95-09 *Monitoring and Training of Shippers and Carriers of Radioactive Materials,* November 3, 1995, 4 pages plus 2 pages of attachments.

NRC GL 95-10 *Relocation of Selected Technical Specifications Requirements Related to Instrumentation,* December 15, 1995, 5 pages plus 6 pages of attachments.

NRC GL 96-02 *Reconsideration of Nuclear Power Plant Security Requirements Associated with an Internal Threat,* February 13, 1996, 5 pages plus one-page attachment.

NRC GL 96-03 *Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits,* January 31, 1996, 5 pages plus 7 pages of attachments.

NRC Bulletins

NRC B 95-02 *Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in*

Suppression Pool Cooling Mode, October 17, 1995, 7 pages plus one-page attachment.

NRC B 96-01 *Control Rod Insertion Problems,* March 8, 1996, 6 pages plus one-page attachment.

NRC Information Notices

NRC IN 95-45 *American Power Service Falsification of American Society for Nondestructive Testing (ASNT) Certificates,* October 4, 1995, 2 pages plus one-page attachment.

NRC IN 95-46 *Unplanned, Undetected Release of Radioactivity from the Exhaust Ventilation System of a Boiling Water Reactor,* October 6, 1995, 3 pages plus one-page attachment.

NRC IN 95-47, Revision 1 *Unexpected Opening of a Safety/Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage,* November 30, 1995, 5 pages plus one-page attachment.

NRC IN 95-48 *Results of Shift Staffing Study,* October 10, 1995, 3 pages plus one-age attachment.

NRC IN 95-49 *Seismic Adequacy of Thermo-Lag Panels,* 4 pages plus one-page attachment.

NRC IN 95-50 *Safety Defect in GammaMed 12i Bronchial Catheter Clamping Adapters,* October 30, 1995, 2 pages plus 5 pages of attachments.

NRC IN 95-51 *Recent Incidents Involving Potential Loss of Control of Licensed Material,* October 27, 1995, 5 pages plus 3 pages of attachments.

NRC IN 95-52 *Fire Endurance Test Results for Electrical Raceway Fire Barrier Systems Constructed from 3M Company Interam Fire Barrier Materials,* November 14, 1995, 4 pages plus 4 pages of attachments.

NRC IN 95-53 *Failures of Main Steam Isolation Valves as a Result of Sticking Solenoid Pilot Valves,* December 1, 1995, 4 pages plus 2 pages of attachments.

NRC IN 95-54 *Decay Heat Management Practices During Refueling Outages,* December 1, 1995, 4 pages plus 2 pages of attachments.

NRC IN 95-55 *Handling Uncontained Yellowcake Outside of a Facility Processing Circuit,* December 6, 1995.

NRC IN 95-56 *Shielding Deficiency in Spent Fuel Transfer Canal at a Boiling-Water Reactor,* December 11, 1995, 3 pages plus 2 pages of attachments.

NRC IN 95-57 *Risk Impact Study Regarding Maintenance During Low-Power Operation and Shutdown,* December 13, 1995, 2 pages plus one-page attachment.

NRC IN 95-58 10CFR34.20—*Final Effective Date,* December 18, 1995, 4 pages plus one-page attachment.

NRC IN 96-01 *Potential for High Post-Accident Closed-Cycle Cooling Water Temperatures to Disable Equipment Important to Safety,* January 3, 1996.

NRC IN 96-02 *Inoperability of Power-Operated Relief Valves Masked by Down-Stream Indications During Testing,* January 5, 1996, 4 pages plus 2 pages of attachments.

NRC IN 96-03 *Main Steam Safety Valve Setpoint Variation as a Result of Thermal Effects,* January 5, 1996, 3 pages plus one-page attachment.

NRC IN 96-04 *Incident Reporting Requirements for Radiography Licensees,* January 10, 1996, 4 pages plus 4 pages of attachments.

NRC IN 96-05 *Partial Bypass of Shutdown Cooling Flow from the Reactor Vessel,* January 18, 1996, 3 pages plus 2 pages of attachments.

NRC IN 96-06 *Design and Testing Deficiencies of Tornado Dampers at Nuclear Power Plants,* January 25, 1996, 3 pages plus one-page attachment.

NRC IN 96-07 *Slow Five Percent Scram Insertion Times Caused by Viton Diaphragms in Scram Solenoid Pilot Valves,* January 26, 1996, 3 pages plus one-page attachment.

NRC IN 96-08 *Thermally Induced Pressure Locking of a High Pressure Coolant Injection Gate Valve,* February 5, 1996, 4 pages plus one-page attachment.

NRC IN 96-09 *Damage in Foreign Steam Generator Internals,* February 12, 1996, 4 pages plus one-page attachment.

NRC IN 96-10 *Potential Blockage by Debris of Safety System Piping Which is Not Used During Normal Operation or Testing During Surveillances,* February 13, 1996, 4 pages plus 2 pages of attachments.

NRC IN 96-11 *Ingress of Demineralizer Resins Increases Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations,* February 14, 1996, 4 pages plus one-page attachment.

NRC IN 96-12 *Control Rod Insertion Problems,* February 15, 1996, 3 pages plus one-page attachment.

NRC IN 96-13 *Potential Containment Leak Paths Through Hydrogen Analyzers,* February 26, 1996, 3 pages plus one-page attachment.

NRC IN 96-14 *Degradation of Radwaste Facility Equipment at Millstone Nuclear Power Station, Unit 1,* March 1, 1996, 2 pages plus one-page attachment.

NRC IN 96-15 *Unexpected Plant Performance During Performance of New Surveillance Tests,* March 8, 1996, 2 pages plus one-page attachment.

NRC IN 96-16 *BWR Operation with Indicated Flow Less Than Natural Circulation,* March 14, 1996, 4 pages plus one-page attachment.

NRC IN 96-17 *Reactor Operation Inconsistent with the Updated Final Safety Analysis Report,* March 18, 1996, 2 pages plus 15 pages of attachments.

NRC IN 96-18 *Compliance with 10 CFR Part 20 for Airborne Thorium,* March 25, 1996, 5 pages plus one-page attachment.

Other Operations Reports

These are other reports issued by various organizations in the United States dealing with power-reactor operations activities. Most of the NRC publications (NUREG series documents) can be ordered from the Superintendent of Documents, U.S. Government Printing Office (GPO), P.O. Box 37082, Washington, DC 20013. NRC draft copies of reports are available free of charge by writing the NRC Office of Administration (ADM), Distribution and Mail Services Section, Washington, DC 20555. A number of these reports can also be obtained from the NRC Public Document Room. Specify the report number when ordering. Telephone orders can be made by contacting the PDR at (202) 634-3273.

Many other reports prepared by U.S. Government laboratories and contractor organizations are available from the U.S. Department of Commerce, Technology Administration, National Technical Information Service, Springfield, VA 22161, and/or DOE Office of Scientific and Technical Information (OSTI), P.O. Box 62, Oak Ridge, TN 37831. Reports available through one or more of these organizations are designated with the appropriate information (i.e., GPO, PDR, NTIS, and OSTI) in parentheses at the end of the listing, followed by the price, when available.

NUREG-0090, Vol. 18, No. 2 *Report to Congress on Abnormal Occurrences for April–June 1995,* October 1995, 18 pages (GPO).

NUREG-0090, Vol. 18, No. 3 *Report to Congress on Abnormal Occurrences for July–September 1995,* February 1996, 14 pages (GPO).

NUREG-0713, Vol. 16 *Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities, 1994,* Twenty-Seventh Annual Report, M. L. Thomas and D. Hagemeyer, January 1996, 300 pages (GPO).

NUREG/CR-2850, Vol. 14 *Dose Commitments Due to Radioactive Releases from Nuclear Power Plant Sites in 1992,* R. L. Aaberg and D. A. Baker, Pacific Northwest Lab., WA, March 1996, 183 pages (GPO).

NUREG/CR-2907, Vol. 14 *Radioactive Materials Released from Nuclear Power Plants, Annual Report 1993,* J. Tichler et al., Brookhaven National Lab., NY, December 1995, 320 pages (GPO).

NUREG/CR-2850, Vol. 13 *Dose Commitments Due to Radioactive Releases from Nuclear Power Plant Sites in 1991,* D. A. Baker, Pacific Northwest Lab., WA, April 1995, 175 pages (GPO).

NUREG/CR-6339 *Aging Assessment of Westinghouse PWR and General Electric BWR Containment Isolation Functions,* B. S. Lee et al., Brookhaven National Lab., NY, March 1996, 135 pages (GPO).

NUREG/CR-6442 *Evidence of Aging Effects on Certain Safety-Related Components,* H. L. Magleby et al., Idaho National Engineering Lab., ID, January 1996, 65 pages (GPO).

NRC Office for Analysis and Evaluation of Operational Data

The NRC Office for Analysis and Evaluation of Operational Data (AEOD) is responsible for the review and assessment of commercial nuclear power plant operating experience. AEOD publishes a number of reports, including case studies, special studies, engineering evaluations, and technical reviews. Individual copies of these reports may be obtained from the NRC Public Document Room or from the GPO.

AEOD/E96-01 *Motor-Operated Valve Key Failures,* C. Hsu, March 1996, 45 pages (GPO).

AEOD/T96-01 *AEOD Technical Reports by Category,* S. Israel, March 1996, 45 pages (GPO).

Special Report—Emergency Diesel Generator Power System Reliability 1987–1993 INEL-95-0035, G. M. Grant et al., February 1996, 185 pages (GPO).

DOE- and NRC-Related Items

NUREG-1530 *Reassessment of NRC's Dollar Per Person-Rem Conversion Factor Policy,* December 1995, 15 pages (GPO).

NUREG/CR-5884, Vols. 1&2 *Revised Analyses of Decommissioning for the Reference Pressurized Water Reactor Power Station. Effects of Current Regulatory and Other Considerations on the Financial Assurance Requirements of the Decommissioning Rule and on Estimates of Occupational Radiation Exposure, Main Report and Appendices,* G. J. Konzek et al., Pacific Northwest Lab., WA, November 1995, 425 pages (GPO).

NUREG/CR-6054 *Estimating Pressurized Water Reactor Decommissioning Costs. A User's Manual for the PWR Cost Estimating Computer Program (CECP) Software, Final Report,* M. C. Bierschbach, Pacific Northwest Lab., WA, November 1995, 140 pages (GPO).

NUREG/CR-6239, Vols. 1&2 *Survey of Strong Motion Earthquake Effects on Thermal Power Plants in California with Emphasis on Piping Systems, Main Report and Appendices,* J. D. Stevenson, Oak Ridge National Lab., TN, November 1995, 225 pages (GPO).

NUREG/CR-6340 *Aging Assessment of Surge Protective Devices in Nuclear Power Plants,* J. F. Davis et al., Brookhaven National Lab., NY, January 1996, 165 pages (GPO).

NUREG/CR-6343 *On-Line Testing of Calibration of Process Instrumentation Channels in Nuclear Power*

Plants, Phase II Final Report, H. M. Hashemian, Analysis and Measurement Services Corp., TN, November 1995, 305 pages (GPO).

NUREG/CR-6349 *Cost-Benefit Considerations in Regulatory Analysis,* V. Mubayi et al., Brookhaven National Lab., NY, October 1995, 130 pages (GPO).

NUREG/CR-6353 *Comments Received on Proposed Rule on Radiological Criteria for Decommissioning and Related Documents,* G. Page et al., Advanced Systems Technology Inc., MD, March 1996, 155 pages (GPO).

NUREG/CR-6382 *Comparisons of ASTM Standards Cited in the NRC Standard Review Plan, NUREG-0800, and Related Documents,* A. R. Ankrum et al., Pacific Northwest Lab., WA, October 1995, 100 pages (GPO).

NUREG/CR-6358, Vols. 1&2 *Assessment of United States Industry Structural Codes and Standards for Application to Advanced Nuclear Power Reactors, Final Report and Appendices,* T. M. Adams and J. D. Stevenson, Stevenson and Associates, OH, October 1995, 300 pages (GPO).

NUREG/CR-6385 *Comparison of ANS, ASME, AWS and NFPA Standards Cited in the NRC Standard Review Plan, NUREG-0800, and Related Documents,* A. R. Ankrum et al., Pacific Northwest Lab., WA, November 1995, 115 pages (GPO).

NUREG/CR-6386 *Comparison of ANSI Standards Cited in the NRC Standard Review Plan, NUREG-0800, and Related Documents,* A. R. Ankrum, Pacific Northwest Lab., WA, November 1995, 100 pages (GPO).

NUREG/CR-6396 *Examples, Clarifications, and Guidance on Preparing Requests for Relief from Pump and Valve Inservice Testing Requirements,* C. B. Ranson and R. S. Hartley, Idaho National Engineering Lab., ID, February 1996, 168 pages (GPO).

NUREG/CR-6422 *Power Excursion Analysis for High Burnup Cores,* D. J. Diamond et al., Brookhaven National Lab., NY, February 1996, 69 pages (GPO).

NUREG/CR-6424 *Report on Aging of Nuclear Power Plant Reinforced Concrete Structures,* D. J. Naus et al., Oak Ridge National Lab., TN, March 1996, 271 pages (GPO).

NUREG/CR-6425 *Impact of Structural Aging on Seismic Risk Assessment of Reinforced Concrete Structures in Nuclear Power Plants,* B. Ellingwood and J. Song, Oak Ridge National Lab., TN, March 1996, 70 pages (GPO).

NUREG/CR-6430 *Software Safety Hazard Analysis,* J. D. Lawrence, Lawrence Livermore Lab., CA, February 1996, 80 pages (GPO).

NUREG/CR-6435 *An Analysis of the Impacts of Economic Incentive Programs on Commercial Nuclear Power Plant Operations and Maintenance Costs,* D. C. Kavanaugh et al., Pacific Northwest Lab., WA, February 1996, 31 pages (GPO).

Other Items

IAEA-RDS-1/15 *Energy, Electricity and Nuclear Power Estimates for the Period up to 2015—July 1995 Edition,* International Atomic Energy Agency (IAEA), November 1995, 53 pages (available from UNIPUB, 4611-F Assembly Drive, Lanham, MD 20706-4391).

IAEA-RDS-3/9 *Nuclear Research Reactors in the World—December 1995 Edition,* IAEA, January 1996, 132 pages (available from UNIPUB).

STI/PUB/971 *Environmental Impact of Radioactive Releases,* IAEA, January 1996, 874 pages (available from UNIPUB).

STI/PUB/984 *External Man-Induced Events in Relation to Nuclear Power Plants: A Safety Guide,* IAEA, January 1996, 70 pages (available from UNIPUB).

STI/PUB/993 *Direct Methods for Measuring Radionuclides in the Human Body: A Safety Practice,* IAEA, March 1996, 110 pages (available from UNIPUB).

STI/PUB/994 *Human Reliability Analysis in Probabilistic Safety Assessment for Nuclear Power Plants,* IAEA, January 1996, 99 pages (available from UNIPUB).

STI/PUB/999 *Operating Experience with Nuclear Power Stations in Member States in 1994,* IAEA, January 1996, 864 pages (available from UNIPUB).

STI/PUB/1000 *Radiation Protection and the Safety of Radiation Sources: A Safety Fundamental,* IAEA, February 1996, 24 pages (available from UNIPUB).

NCRP Report 122 *Use of Personnel Monitors to Estimate Effective Dose Equivalent and Effective Dose to Workers for External Exposure to Low-LET Radiation,* National Council on Radiation Protection and Measurements (NCRP), MD, December 1995, 64 pages (available from NCRP Publications, 7910 Woodmont Ave., Suite 800, Bethesda, MD 20814-3095).

NCRP Report 123I *Screening Models for Releases of Radionuclides to Atmosphere, Surface Water, and Ground,* NCRP, January 1996, 316 pages (available from NCRP).

NCRP Report 123II *Screening Models for Releases of Radionuclides to Atmosphere, Surface Water, and Ground—Work Sheets,* NCRP, January 1996, 204 pages (available from NCRP).

EPRI-TR-105909 *Generic Framework for Application of Revised Accident Source Term to Operating Plants,* D. E. Leaver and J. Metcalf, Electric Power Research Inst. (EPRI), CA, November 1995, available from NRC/PDR.

REGULATORY GUIDES

To expedite the role and function of the NRC, its Office of Nuclear Regulatory Research prepares and maintains a file of Regulatory Guides that define much of the basis for the licensing of nuclear facilities. These Regulatory Guides are divided into 10 divisions as shown in Table 1.

Table 1 Regulatory Guides

Division 1	Power Reactor Guides
Division 2	Research and Test Reactor Guides
Division 3	Fuels and Materials Facilities Guides
Division 4	Environmental and Siting Guides
Division 5	Materials and Plant Protection Guides
Division 6	Product Guides
Division 7	Transportation Guides
Division 8	Occupational Health Guides
Division 9	Antitrust and Financial Review Guides
Division 10	General Guides

Single copies of the draft guides may be obtained from NRC Distribution Section, Division of Information Support Services, Washington, DC 20555. Draft guides are issued free (for comment) and licensees receive both draft and final copies free; others can purchase single copies of active guides by contacting the U.S. Government Printing Office (GPO), Superintendent of Documents, P.O. Box 37082, Washington, DC 20013. Costs vary according to length of the guide. Of course, draft and active copies will be available from the NRC Public Document Room, 1717 H Street, NW, Washington, DC, for inspection and copying for a fee.

Revisions in these rates will be announced as appropriate. Subscription requests should be sent to the National Technical Information Service, Subscription Department, Springfield, VA 22161. Any questions or comments about the sale of regulatory guides should be directed to the Chief, Document Management Branch, Division of Technical Information and Document Control, Nuclear Regulatory Commission, Washington, DC 20555.

Actions pertaining to specific guides (such as issuance of new guides, issuance for comment, or withdrawal), which occurred during the reporting period, are listed.

Division 1 Power Reactor Guides

1.152 (Rev. 1) *Criteria for Digital Computers in Safety Systems of Nuclear Power Plants,* January 1996.

1.153 (Draft, Proposed Rev. 1) *Criteria for Safety Systems,* November 1995.

Division 5 Materials and Plant Protection Guides

5.015 (Draft, Proposed Rev. 1) *Tamper-Indicating Seals for Protection and Control of Special Nuclear Materials,* January 1996.

Division 8 Occupational Health Guides

8.29 (Rev. 1) *Instruction Regarding Risks from Occupational Radiation Exposure,* February 1996.

8.37 (Proposed Rev. 1) *Constraints for Air Effluents for Licensees Other Than Power Reactors,* December 1995.

NUCLEAR STANDARDS

Standards pertaining to nuclear materials and facilities are prepared by many technical societies, international organizations, the U.S. Department of Energy, etc. When standards prepared by a technical society are submitted to the American National Standards Institute (ANSI), they are assigned ANSI standard numbers, although they may also contain the identification of the originating organization and be sold by that organization as well as by ANSI.

Editor's Note: Normally, we would list here the most significant nuclear standards actions taken by organizations from October 1995 through March 1996. Regrettably, this list was unavailable at the time this issue was sent to the printer. We regret any inconvenience this may cause.

Proposed Rule Changes as of Dec. 31, 1995[a,b]

(Changes Since the Previous Issue of *Nuclear Safety* Are Indicated by Shaded Areas)

Number of part to be changed	Date published for comment	Date comment period expired	Date published; date effective	Topic or proposed effect	Current action and/or comment, *Federal Register* volumes and page numbers
10 CFR 2	9-29-93	11-15-93		Informal hearing procedures for materials licensing adjudications	Published for comment in 58:187 (50858)
10 CFR 2	8-23-94; 9-27-94; 11-28-94	10-24-94; 12-28-94		Reexamination of the NRC enforcement policy	Published for comment in 59:162 (43298); correction in 59:171 (46004); expanded scope in 59:186 (49215); revised in 59:227 (60697)
10 CFR 2	3-28-95	6-12-95		Petition for rulemaking; procedure for submission	Published for comment in 60:059 (15878)
10 CFR 2, 50, 51	7-20-95	10-18-95		Decommissioning of nuclear power reactors	Published for comment in 60:139 (37374)
10 CFR 9			12-13-95; 1-12-96	Revision of specific exemptions under the privacy act	Published for comment in 60:143 (38282); final rule in 60:239 (63897)
10 CFR 19, 20			7-13-95; 8-14-95	Radiation protection requirements; amended definitions and criteria	Published for comment in 59:023 (5132); final rule in 60:134 (36038)
10 CFR 20	6-18-93	8-15-93; 9-20-93		Radiological criteria for decommissioning of NRC-licensed facilities; generic environmental impact statement (GEIS) for rulemaking, notice of intent to prepare a GEIS and to conduct a scoping process	Published for comment in 58:116 (33570); comment period extended in 58:154(42882)
10 CFR 20	2-2-94	3-11-94		Radiological criteria for decommissioning of NRC-licensed facilities; enhanced participatory rulemaking, availability of the staff's draft of the rule	Published for comment in 59:022 (4868)
10 CFR 20	2-25-94	5-26-94		Disposal of radioactive material by release into sanitary sewer systems	Advanced notice of proposed rulemaking published in 59:038 (9146)
10 CFR 20	12-13-95	3-12-96		Constraint level for air emissions of radionuclides	Published for comment in 60:239 (63984)
10 CFR 20, 30,40,50, 51,70,72	8-22-94	12-20-94; 1-20-95		Radiological criteria for decommissioning	Published for comment in 59:161, Part III (43200); comment period extended in 59:236 (63733); schedule extension in 60:151 (40117)

Proposed Rule Changes as of Dec. 31, 1995 (Continued)

Number of part to be changed	Date published for comment	Date comment period expired	Date published; date effective	Topic or proposed effect	Current action and/or comment, *Federal Register* volumes and page numbers
10 CFR 20, 30,40,61, 70,72	12-28-94	3-28-95		Termination or transfer of licensed activities: recordkeeping requirements	Published for comment in 59:248 (66814)
10 CFR 20 10 CFR 35	6-15-94	8-29-94		Criteria for the release of patients administered radioactive material	Published for comment in 59:114 (30724)
10 CFR 20 10 CFR 35			9-20-95; 10-20-95	Medical administration of radiation and radioactive materials	Published for comment in 60:016 (4872); final rule in 60:182 (48623)
10 CFR 21			9-19-95; 10-19-95	Procurement of commercial grade items by nuclear power plant licensees	Published for comment in 59:204 (53372); final rule in 60:181 (48369)
10 CFR 26	5-11-94	9-9-94		Consideration of changes to fitness-for-duty (FFD) requirements	Published for comment in 59:090 (24373)
10 CFR 30, 40, 70	9-8-95	10-10-95		One-time extension of certain byproduct, source, and special nuclear materials licenses	Published for comment in 60:174 (46784)
10 CFR 30, 40, 70, 72			7-26-95; 11-24-95	Clarification of decommissioning funding requirements	Published for comment in 59:119 (32138); final rule in 60:143 (38235)
10 CFR 34 10 CFR 150	2-28-94	5-31-94		Licenses for radiography and radiation safety requirements for radiographic operations	Published for comment in 59:039 (9429)
10 CFR 35	11-3-94	3-3-95		Request for comments regarding potential modifications of NRC's therapy regulations	Published for comment in 59:212 (55068)
10 CFR 50	6-28-93; 4-14-95	9-13-93; 7-13-95		Production and utilization facilities; emergency planning and preparedness-exercise requirements	Published for comment in 58:122 (34539); published for comment in 60:072 (19002)
10 CFR 50	1-7-94	3-24-94; 4-25-94		Codes and standards for nuclear power plants; subsection IWE and subsection IWL	Published for comment in 59:005 (979); comment period extended in 59:059 (4373)
10 CFR 50	9-19-94	12-5-94		Steam generator tube integrity for operating nuclear power plants	Published for comment in 59:180 (47817)

(Table continues on the next page.)

Proposed Rule Changes as of Dec. 31, 1995 (Continued)

Number of part to be changed	Date published for comment	Date comment period expired	Date published; date effective	Topic or proposed effect	Current action and/or comment, *Federal Register* volumes and page numbers
10 CFR 50			12-19-95; 1-18-96	Fracture toughness requirements for light water reactor pressure vessels	Published for comment in 59:191 (50513); final rule in 60:243 (65456)
10 CFR 50	10-19-94; 10-25-94; 1-18-95	1-3-95; 2-3-95		Shutdown and low-power operations for nuclear power reactors	Published for comment in 59:201 (52707); correction in 59:205 (53613); comment period extended in 60:011 (3579)
10 CFR 50			9-26-95; 10-26-95	Primary reactor containment leakage testing for water-cooled power reactors	Published for comment in 60:034 (9634); final rule in 60:186 (49495)
10 CFR 50			7-19-95; 8-18-95	Technical specifications	Published for comment in 59:181 (48180); final rule in 60:138 (36953)
10 CFR 50	9-14-95	11-28-95		Receipt of a petition for rulemaking filed by the Nuclear Energy Institute	Published for comment in 60:178 (47716)
10 CFR 50	11-27-95	2-12-96		Receipt of petition for rulemaking by Peter G. Crane	Published for comment in 60:227 (58256)
10 CFR 50, 52, 100	10-20-92	2-17-93; 3-24-93; 6-1-93; 2-14-95; 5-12-95		Reactor site criteria, including seismic and earthquake engineering criteria for nuclear power plants and proposed denial of petition for rulemaking from Free Environment, Inc., et al.	Published for comment in 57:203 (47802); comment period extended in 58:002 (271); extended again in 58:057 (16377); extended again in 59:199 (52255); extended again in 60:026 (7467); extension deadline set 60:039 (10810)
10 CFR 50, 70, 72			10-16-95; 11-15-95	Physical security plan format changes	Published for comment in 60:073 (19170); final rule in 60:199 (53505)
10 CFR 51	9-17-91	12-16-91; 3-16-92; 9-8-94		Environmental review for renewal of operating licenses	Published for comment in 56:180 (47016); comment period extended in 56:228 (59898); supplemental proposed rulemaking in 59:141 (37724)
10 CFR 52	11-3-93	1-3-94		Rulemakings to grant standard design certification for evolutionary light water reactor designs	Advance notice of proposed rulemaking published in 58:211 (58664)

Proposed Rule Changes as of Dec. 31, 1995 (Continued)

Number of part to be changed	Date published for comment	Date comment period expired	Date published; date effective	Topic or proposed effect	Current action and/or comment, *Federal Register* volumes and page numbers
10 CFR 52	4-7-95	8-7-95		Standard design certification for the U.S. Advanced Boiling Water Reactor design	Published for comment in 60:067 (17902)
10 CFR 52	4-7-95	8-7-95		Standard design certification for the System 80+ design	Published for comment in 60:067 (17924)
10 CFR 60	7-9-93	10-7-93		Disposal of high-level radioactive wastes in geologic repositories; investigation and evaluation of potentially adverse conditions	Published for comment in 58:130 (36902)
10 CFR 60	3-22-95	6-20-95		Disposal of high-level radioactive wastes in geologic repositories; design basis events	Published for comment in 60:055 (15180)
10 CFR 60, 72, 73, 75	8-15-95	11-13-95		Safeguards for spent nuclear fuel or high-level radioactive waste	Published for comment in 60:157 (42079)
10 CFR 61	7-18-95	10-3-94; 12-2-94		Land ownership requirements for low-level waste sites	Published for comment in 59:148 (39485); comment period extended in 59:202 (52941); withdrawal of advanced notice of proposed rulemaking in 60:137 (36744)
10 CFR 71			9-28-95; 4-1-96	NRC revising regulations governing the transportation of radioactive material	Final rule in 60:188 (50248)
10 CFR 73			9-7-95; 10-10-95	Changes to nuclear power plant security requirements associated with containment access control	Published for comment in 60:090 (24803); final rule in 60:173 (46497)
10 CFR 110			7-21-95; 8-21-95	Regulations establishing specific licensing requirements for import and export of incidental radioactive material	Final rule in 60:140 (37556)

[a]NRC petitions for rulemaking are not included here, but quarterly listings of such petitions can be obtained by writing to Division of Rules and Records, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Quarterly listings of the status of proposed rules are also available from the same address.

[b]Proposed rules for which the comment period expired more than 2 years prior to the start of the period currently covered without any subsequent action are dropped from this table. Effective rules are removed from this listing in the issue after their effective date is announced.

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The Nuclear Community and the Public: Cognitive and Cultural Influences on Thinking About Nuclear Risk

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Twenty-Third Water Reactor Safety Information Meeting

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Analysis of a PWR LBLOCA Without SCRAM

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Vulnerability of Multiple-Barrier Systems

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A Study of Wet Catalytic Oxidation of Radioactive Spent Ion Exchange Resin by Hydrogen Peroxide

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A Comparison Study and Resolution of Differences Between Emergency Response and Safety Analysis Codes Used at the Savannah River Site

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