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ANALYSES OF PLUME FORMATION, AEROSOL AGGLOMERATION AND RAINOUT FOLLOWING CONTAINMENT FAILURE

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(2) The aerosol agglomeration calculations would have to be performed for the complete blowdown process, not just the start-of-blowdown conditions emphasized in the present study. This would require a model development effort because current models are limited to constant upstream conditions. Conditions become substantially more favorable to rainout as blowdown proceeds.

(3) Aerosol agglomeration calculations would have to be performed for a considerably wider range of containment, thermal-hydraulic, and meteorological conditions than has been treated here (in the present treatment, aerosol calculations were performed for only three of the 25 cases of the thermal-hydraulic/meteorological sensitivity study).

(4) Results of tasks 1 through 3 would have to be applied in the context of risk-dominant sequences at specific plants, including site-specific meteorology.

Should the result of the first task above indicate that the degree of agglomeration falls somewhere in the lower portion of the present uncertainty band, it is likely that tasks 2 and 3 would confirm that rainout is of no direct importance to risk. If the first task should show that agglomeration falls in the upper portion of the present uncertainty band (e.g., is comparable to that calculated using the WETJET base case parameters), identification of the combinations of parameters yielding significant rainout from tasks 2 and 3 might lead to a significant, direct reduction in risk estimates, but this result is certainly not assured.

8.2 The Steam Generator Tube Rupture Event at Ginna

In 1982, a steam generator tube rupture event was reported at the R.F. Ginna Nuclear Power Plant. At the time of the generator tube rupture, and for the entire period of interest, there was a wind speed of about 6 m/s (measured at a height of 10 m above the ground). The atmospheric stability was neutral, the temperature was about -12°C, and light snow was falling. During the incident, primary system coolant was released to the atmosphere and a snowfall from the plume was reported.

An analysis of the event is given in Ramsdell et al., [Ra83]. In their evaluation, they model the release as an isenthalpic expansion across the safety vents. However, they note their flow velocities are unrealistically high, and the flow in the vents is likely to be sonic. Also, they assume that the moisture in the plume does not enhance washout, but they note that this assumption is probably not good since the Ginna plume probably contained a large water excess; the excess water would probably

condense and fall out of the plume near the release point, and the surface deposition near the release point would probably be higher than their results indicate while the distant surface concentrations would probably be lower.

During the later stages of the accident, the main steam line from the generator filled with water. As a result, water rather than steam was vented at that time. It is believed that much of the the contamination released through the safety valve was associated with the fraction of the effluent released as liquid water. In the Ramsdell [Ra83] analysis, they conclude that the liquid leaving the vents will fall near (within 100 m) the source.

The specific conditions that were present during the Ginna steam generator tube rupture are not addressed in this report. However, the modeling techniques developed for this investigation could possibly be extended to perform an analysis of the Ginna incident at least in the early stages of the release before the water was released. The Ramsdell [Ra83] report provides information about the plume trajectory, the flow velocities in the plume and the plume water content. With this information, the model developed for this investigation could be used to estimate the additional washout that occurred due to condensation in the plume. Application of the current model to the later stages of the incident would be more difficult. The models developed here do not apply to cases where the steam quality is much less than one. They would have to be modified in order to model the the later stages of the event.

8.3 Consequence Modeling Implications

Currently, the transition, from a highly pressurized containment state to a low pressure plume state is not modeled in containment response codes and is treated parametrically in consequence codes.

If the transition from the containment state to the plume state occurs through a jet state as described in this study, not only is there the potential for rainout, but the initial conditions for a typical consequence calculation could be altered. Even in cases where droplets of rainout size are not produced, a change in the size of the residual solid particles (after evaporation) can occur. Also, a change in the the thermodynamic state due to cooling and condensation in the jet can occur. Finally, the initial height of the plume may be affected by the expansion that occurs during the jet phase and aerodynamic forces created by the jet that may pull the jet toward the ground.

These changes in the initial conditions may very well produce only small perturbations in the consequence calculations, however. For instance, an increase in the diameter of the

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Soviet Medical Response to the Chernobyl Nuclear Accident

Roger E. Linnemann, MD

The nuclear accident at Chernobyl was the worst in the history of nuclear power. It tested the organized medical response to mass radiation casualties. This article reviews the Soviet response as reported at the 1986 postaccident review meeting in Vienna and as determined from interviews. The Soviets used three levels of care: rescue and first aid at the plant site; emergency treatment at regional hospitals; and definitive evaluation and treatment in Moscow. Diagnosis, triage, patient disposition, attendant exposure, and preventive actions are detailed. The United States would be well advised to organize its resources definitively to cope with future nonmilitary nuclear accidents.

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THE NUCLEAR accident at Chernobyl was the worst in the history of nuclear power, and it was the first test of an organized emergency medical response to mass radiation casualties. The present account of the Soviet emergency medical plan is based on my participation in the postaccident review meeting conducted in Vienna from Aug 24 to 29, 1986. The meeting, sponsored by the International Atomic Energy Agency, was conducted by a group of experts in various nuclear disciplines from the Soviet Union. It was the first official Soviet report of the cause and consequences of the Chernobyl accident.[1] The meeting was exceptionally well organized, and the Soviets were exceedingly open and forthright. In addition to the formal presentations, I was able to obtain additional information in personal conversations with Angelina Guskova, MD, a Russian radiologist who is chief of the Institute of Biophysics, Moscow, and with Leonid Ilyin, MD, vice president of the Soviet Academy of Medical Sciences, Moscow. Dr Guskova played a key role in both the emergency medical response and management of patients in the aftermath of the accident.

The main purpose of the present report is to describe the organizational aspects of the Soviet medical response. It should be emphasized that the Soviet report on which the present article is heavily dependent is preliminary.[illegible] The medical data presented in the Soviet report and those data that have subsequently become available through interviews, personal communications, and verbal presentations by Soviet physicians must, of necessity, be carefully interpreted. For example, the particular cytologic techniques used to obtain blood cell counts are not given. Dose estimates derived from such measurements can be highly dependent on the techniques employed. Also, in the present article, the dose unit, gray (rad), is repeated as given in the Soviet report. However, most of the conclusions on individual doses are based on biologic criteria such as vomiting and results of cell cytometry. Properly speaking, doses so derived may be reported only in terms of the unit "roentgen equivalents man" (sievert).

Considering the volume of data gathered by hundreds of medical personnel on thousands of evacuees and patients, the Soviet scientists have done a commendable job in the analysis of data and preparation of the report in the short time between the accident and the meeting in Vienna. Though more formal scientific articles and data analyses are expected from the Soviet medical community, it will be years before detailed, definitive scientific reports on each topic and subtopic reach publication.

The Chernobyl site, located 80 km north of Kiev and 3 km from Pripyat, had four operating reactors, one reactor under construction, and one planned. The four operating units were graphite-moderated, water-cooled reactors, which initially were used almost exclusively in the Soviet Union to generate electricity. Twenty-three of the 44 operating nuclear power plants in the Soviet Union are of the graphite type.[1] In contradistinction, Western countries adopted the water-moderated, water-cooled reactor for commercial use of nuclear power. The moderating medium is important in the optimization of the speed of neutrons necessary to cause fissioning of uranium 235 atoms. The choice of graphite as a moderator was a critical factor in the medical consequences of this accident.

Because of their confidence in the design of the reactor, the Soviets did not enclose the entire unit with a containment structure[2] and had not developed either an off-site emergency plan or employed an off-site monitoring system. However, they do appear to have had a

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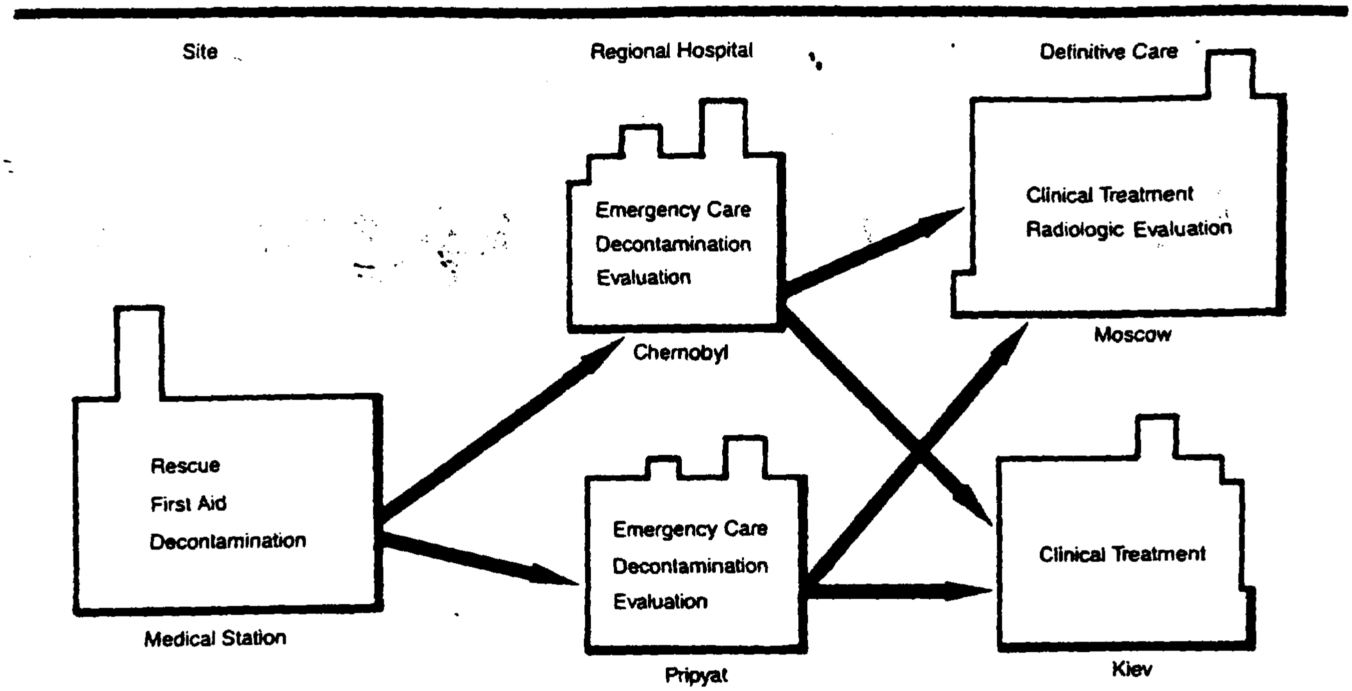

Fig 1.—Soviet plan defined three levels of care: first aid and rescue at plant site, emergency treatment at regional hospitals, and tertiary care at definitive-care centers.

well-organized medical emergency response system for nuclear accidents.

SOVIET MEDICAL PLAN

The emergency medical plan for nuclear accidents in the Soviet Union was developed in the late 1960s. It specified three levels of care (Fig 1): rescue and first aid at the plant site; emergency treatment at regional hospitals; and definitive evaluation and treatment at a specialized center in Moscow.

At the Chernobyl site, the medical station is located in the administration building. This station is normally staffed with physicians, nurses, and technicians. It is equipped with decontamination facilities, a radiobioassay laboratory, survey instrumentation, ambulance transportation, and a holding facility that can be expanded to 115 beds. The regional hospitals are located in Pripyat and Chernobyl and their primary responsibilities are emergency treatment for trauma, decontamination, and the initial evaluation of radiation injuries. The specialized center in Moscow has two components: a multispecialty clinical center at Hospital No. 6, and a radiologic evaluation center at the Institute of Biophysics.

The radiologic evaluation center comprises the following facilities: a radiobioassay laboratory with radiocounting and radiochemistry for blood, tissue, and excreta analysis; a whole-body counter; thyroid uptake counters; radiation cytogenetics; radiopathology; radiobiology; and a multispecialty response team for dispatch to the site of an accident. Dr Guskova and her colleagues established the procedures for the evaluation and triage of patients at both the site and the regional hospitals.

The medical response at Chernobyl was activated in three phases: early, intermediate, and late. The early phase was conducted by site medical personnel on duty at the time of the accident, and by medical support from regional hospitals. The intermediate phase involved the specialized medical-radiologic team from Moscow. The late phase included medical brigades recruited from throughout the Soviet Union to examine evacuees.

INITIAL MEDICAL RESPONSE

At the time of the accident, 1:23 AM, April 26, 1986, 444 workers, including 176 operating staff members and 268 construction workers, were on-site. Hundreds of additional personnel were called in for rescue, plant control, and fire-fighting operations. The three technicians on duty at the medical station were notified within minutes of the accident, and within the first half hour 29 patients were admitted to the medical station. At 2 AM an urgent summons was issued for two surgical and resuscitation teams from Pripyat. One hundred fifteen beds at the medical station were prepared and medical teams were organized to enter the plant for rescue and first aid. A first aid and decontamination station was established at the personnel air lock where all persons enter and exit from the plant. To avoid overwhelming the patient decontamination facilities at the medical station, all patients were stripped of water-soaked, contaminated clothing and given first aid at the personnel air lock before transfer to the medical station. Four more off-site medical teams were summoned, and at 6:40 AM a call was placed for the specialty team in Moscow. Dr Guskova and the team arrived at noon, approximately ten hours after the accident.

The medical plan provided for a two-tiered triage system for handling multiple casualties. The first triage was conducted on site during the hours following the accident. During this phase, victims with serious trauma and a history of high-level exposure to radiation received the highest priority. The second triage took place over the next few days at the regional hospitals where patients with acute radiation syndrome were sorted for evacuation to definitive care centers.

The immediate and most serious medical problem was thermal skin damage. Of five thermal burn victims, two required intensive care for shock and one died at 6 AM, 5½ hours after the accident. In addition, an employee working in the reactor building at the time of the

accident was presumed to have been killed by the explosion and buried by falling debris.

In the absence of physical dosimetry, identification of radiation casualties was based on whether nausea and vomiting occurred. Plant monitors were destroyed in the explosion. The reason for the unavailability of personnel dosimeters was not reported. Presumably, these dosimeters were either off-scale, contaminated, or lost in the confusion. Anyone exhibiting symptoms entered the medical evaluation system. Vomiting was considered particularly significant because its time of onset after exposure can be qualitatively, and to a certain extent quantitatively, related to dose and prognosis.[4]

Although some of the reported or observed nausea and vomiting was not due to radiation exposure, the triage team assumed that it was, pending confirmation of the diagnosis by further testing. Those workers who did not exhibit mild nausea and vomiting until several hours after exposure were decontaminated and sent home. They were asked to return for further evaluation after the crisis had subsided. If, at this time, hematologic evaluation confirmed a large exposure, they were hospitalized.

Typically, the acute radiation syndrome results in nausea and vomiting that abates within one or two days. Some workers who suffered exposure failed to disclose their symptoms of nausea and vomiting. These workers were not identified as having received a significant exposure until several days later, when medical personnel were able to evaluate asymptomatic as well as temporarily symptomatic patients. After being examined in the hospital, some patients were discharged for outpatient follow-up.

Erythema can also be used as a clinical indication of radiation injury. Early erythema, appearing within hours of exposure, is a threshhold phenomenon that affects the germinal layer of the skin after exposure to approximately 12 to 20 Gy (1200 to 2000 rad). It peaks after approximately 24 hours and fades over the next few days.[5] This early erythema is a result of damage to capillaries that are dilated and have increased permeability.[6] Fission products emit medium- to high-energy beta radiations that can penetrate and damage the dermis. Very early erythema in the Chernobyl victims, however, was most likely due to thermal radiation, not beta radiation. Even so, in the attempt to sort out the victims in the first few days, erythema was considered to be suggestive of high-level beta radiation exposure until further evaluation ruled this out. Also, early thermal erythema can usually be distinguished from ionizing radiation injury because it is painful, whereas the latter is not.

Erythema that appears within 24 hours or less can be caused by total-body exposure to penetrating gamma rays and indicates a lethal or near-lethal dose (6 Gy [600 rad]). Skin effects seen in fire fighters and rescue personnel resulted when clothing became soaked with radioactive steam and water as workers attempted to cool the burning graphite with streams of water. The ambient gamma exposures near the reactor were as high as 1 to 1.5 Gy/h (100 to 150 rad/h) (Angelina Guskova, MD, oral communication, August 1986). One Soviet physicist who peered over the wall of the damaged reactor received a dose of 2.5 to 3.0 Gy (25 to 30 rad) in a few minutes (*Philadelphia Inquirer*, April 10, 1986, p 1); this would indicate exposure rates on the order of 4 to 5 Gy/h (400 to 500 rad/h). Skin contamination on some of the fire fighters from water-soaked clothing resulted in near total-body radiation burns that developed over the next few weeks.

The heavy patient load at the medical station in the first hours after the accident precluded complete patient decontamination on-site. After undergoing one or two attempts at decontamination, patients were referred to the regional hospitals. In some cases, contaminated patients were sent to Hospital No. 6 in Moscow (Angelina Guskova, MD, oral communication, August 1986). Within the first 12 hours before the arrival of the specialty team from Moscow, a total of 130 patients had been referred to hospitals in Pripyat and Chernobyl.

When the team of specialists (dermatologists, hematologists, radiobiologists, and physicists) arrived from Moscow, they continued to sort radiation casualties at regional hospitals for evacuation to definitive-care centers. Mucositis, diarrhea, and fever were also considered to be evidence of severe radiation exposure. Complete blood cell counts, including platelet counts, were performed every few hours on hospitalized patients. A rapid drop in both neutrophil and lymphocyte counts in the first few days indicated exposures in the lethal range.[7] Of the 350 patients evaluated for acute radiation syndrome in the first few days, 203 were transferred to clinical centers in Moscow and Kiev. Although high levels of internal contamination were suspected, evaluation and treatment for internal contamination were deferred to definitive-care centers. However, blood samples were obtained on the first day and analyzed for sodium 24 content. None was found, indicating the absence of exposure to neutrons.

DEFINITIVE EVALUATION

Radiologic evaluation continued in Moscow and Kiev. Thousands of white blood cell counts, as many as three and four per patient per day, were performed. In addition, 154 radiation cytogenetic studies were completed in the first three weeks. Electron spin resonance (ESR) was performed on the tooth enamel of four patients who died. The preliminary dose estimates from this procedure agreed well (± 20%) with dose estimates from cytogenetic and neutrophil profiles (Angelina Guskova, MD, oral communication, August 1986). Based on cytogenetic dose estimates, which indicated uniform total-body irradiation in almost all cases, a profile of the neutrophil count was predicted in each of the 115 patients in Moscow. Figure 2 illustrates this predictive curve for a representative patient who received 3.5 Gy (350 rad), compared with the actual neutrophil counts plotted over time. Agreement was excellent.

In Moscow and Kiev, in vivo and in vitro analyses were performed for internal contamination. The major contaminants identified are as follows: iodine 131 and 132; cesium 134 and 137; niobium 95; cerium 144; ruthenium 103 and 106; and plutonium 239. Of these, cesium and iodine isotopes predominated and accounted for 90% of the absorbed dose contributed by internal radionuclides. All patients had significant internal contamination. Dr Guskova believed, however, that the contribution from internal contamination to the total dose was relatively important in only two patients. In one patient, the total-body dose from iodine 131 and 132 and cesium 134 and 137 was estimated to be 4 Gy (400 rad), and in the other patient it was 1.5 Gy (150 rad).[8,9] Both patients also suffered from large external gamma exposures.

On April 28, 1986, urinalysis for alpha radiation activity was performed on ten patients. Three patients showed positive results of 2.0, 0.67, and 0.1 nCi/mL (74, 25, and 3.7 Bq/mL). A diagnostic trial of the drug pentetate calcium trisodium (Pentacine), to accelerate the elimination of plutonium, was unsuccessful.[8,9] Dr Guskova stated that Prussian blue was used to accelerate the elimination of cesium by way of the gastrointestinal tract, but to no avail. As the results of the radiologic evaluations in Moscow and Kiev were analyzed, patients were classified according to previously established dose groups. The

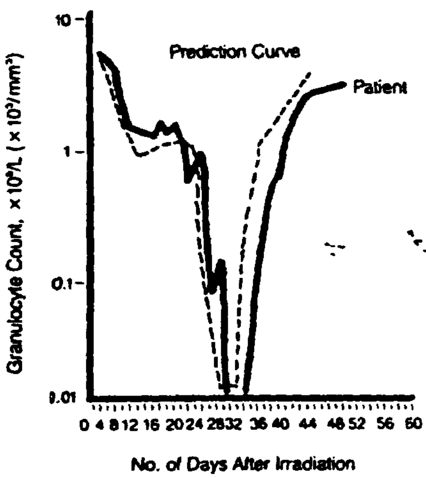

Fig 2—Representative patient who received estimated uniform total-body gamma exposure of 3.3 Gy (330 rad). Broken line indicates prediction of neutrophil profile based on early cytogenetic studies; solid line, actual neutrophil profile as measured in days following accident.

Table 1.—Clinical Response of Established Dose Groups*

Dose Group	Onset of Nausea and Vomiting, h	Latency, d	Lymphocyte Count at 3-6 d, $\times 10^9$/L (per mm^3)	Platelet Count, d	Skin Burns	Dose Range, Gy (rad)
1	2	30	0.6-1.0 (600-1000)	25-28	0	1-2 (100-200)
2	1-2	15-25	0.3-0.5 (300-500)	17-24	Few	2-4 (200-400)
3	0.5-1	8-17	0.1-0.2 (100-200)	10-16	6	4-6 (400-600)
4	0.5	6-8	0.1 (100)	8-10	22	6-16 (600-1600)

*The platelet count reached at the time indicated was 400×10^9/L (40 000/mm^3) or less. The latency is the time between cessation of initial symptoms and the onset of secondary, more severe symptoms.

clinical basis for this classification is presented in Table 1.

MARROW TRANSPLANTATION

Bone marrow transplantation for acute radiation syndrome had been attempted twice before the accident at Chernobyl. In a reactor accident at Vinca, Yugoslavia, in 1958, five workers who received doses greater than 3 Gy (300 rad) were treated with allogenic bone marrow grafting. All but one survived.[8] There was some question, however, as to the effectiveness of the bone marrow transplants, since the patients' own bone marrow showed signs of regeneration after a transitory marrow "take." In the second instance, the transplanted bone marrow was highly effective in decreasing morbidity and enabling survival of the patient. In this case, the patient, who received a total-body dose of 6 Gy (600 rad), received marrow from an identical twin brother on the eighth postirradiation day; by day 21, marrow competence was restored.[9]

A total of 19 marrow transplantations were attempted on the Chernobyl victims. Six were transplants of fetal liver, rich in fetal marrow, and 13 were allogenic bone marrow transplants. All six of the patients who received fetal liver transplants died, and 12 of the 13 allogenic bone marrow transplant recipients died.[10]

The fetal liver transplantations, which are still only under research and development even in the United States, were attempted early by the Soviet physicians in patients with near-fatal skin burns or in patients in whom adequate HLA typing for bone marrow transplantation was impossible because of early loss of lymphocytes as a result of severe exposure to radiation. Fetal liver was used in severely burned patients to avoid risking the life of a donor for bone marrow transplantation. A total of 113 potential donors, close family members in most cases, were evaluated to obtain the 13 bone marrow transplants. Seven of the marrow recipients died between the ninth and 19th days after the transplant from overwhelming skin and gastrointestinal damage before the transplanted marrow could adequately support the patient. In the remaining six bone marrow transplant recipients, only a temporary "take" was observed. Five of these patients died, two of them of complications from the immunosuppression induced for bone marrow transplantation (Angelina Guskova, MD, oral communication, August 1986). In the single survivor, the patient's own bone marrow regenerated after the rejection of the transplanted marrow.

In the opinion of the Soviet physicians, marrow transplant played a minor role in overall patient management. Marrow transplant did not seem to help patients with exposures greater than 10 Gy (1000 rad), in whom the Soviet physicians attributed death to effects on organs other than the bone marrow. Below 5 Gy (500 rad), patients can survive without marrow transplantation. Dr Guskova indicated that, in the future, marrow transplant for the acute radiation syndrome should be used sparingly and only after a long period of observation when it is clinically clear that marrow regeneration is unlikely. Today, with the advances in conservative hematologic care, Dr Guskova believes that patients exposed to as much as approximately 8 Gy (800 rad) have a better chance of survival without marrow transplant. The range of radiation exposure for marrow transplantation is probably between 8 and 12 Gy (800 and 1200 rad). In the final analysis, the patient's clinical condition, not the radiation dose received, should determine the necessity for marrow transplantation.

PATIENT DISPOSITION

Of the 203 victims who were hospitalized, 30 died: one of severe thermal burns within hours of the accident, and 29 of complications of thermal and radiation injury. None of the patients in group 1 (Table 2) died. There were also no skin injuries in this group. There was only one death in group 2, which had few patients with skin injuries. Seven of the 23 patients in group 3 died two to seven weeks following exposure. Six of the seven had suffered severe skin damage. Of the 20 patients in group 4 in Moscow, eight suffered severe skin damage covering 60% to 100% of the body surface, and two had received high levels of internal contamination. Seventeen of the 20 patients in group 4 in Moscow died within ten to 15 days. Two patients in Kiev died within four to ten days. In her presentation in Vienna, Dr Guskova stated that the skin damage in 20 of the victims was in itself life threatening. Robert P. Gale, MD, PhD, Department of Medicine, Division of Hematology/Oncology, UCLA, who assisted the Soviet physicians in bone marrow transplantation, estimated that 50 patients had received more than 5 Gy (500 rad).[11] It is reasonable to assume that the marrow transplants were performed in this group. Since 18 of the 19 transplant recipients died, and the number of deaths among patients hospitalized for acute radiation syndrome was 29, then at least 21 of the patients (42%) who received a dose of 5 Gy (500 rad) or more survived after receiving advanced but conservative hematologic care. This would indicate that treatment can enable more patients to survive larger doses of radiation than was previously believed. However, it is possible that doses received by these patients were actually less than estimated. Further analysis of the data will help clarify this point. The Soviet experience to date indicates that bone marrow transplant

Table 2.—Disposition of Patients With Acute Radiation Syndrome*

Group	Location, No. of Patients: Kiev	Location, No. of Patients: Moscow	No. of Deaths	Dose, Gy (rad)
1	74	31	0	1-2 (100-200)
2	10	43	1	2-4 (200-400)
3	2	21	7	4-6 (400-600)
4	2	20	21	6-16 (600-1600)
Total	88	115	29	...

*A total of 203 patients were admitted for acute radiation syndrome. After initial clinical evaluation and observation, the patients were classified into these four exposure groups. The more severely irradiated patients were treated in the designated center in Moscow.

has a specific but limited use in the treatment of the acute radiation syndrome.

ATTENDANT EXPOSURE

Although the total number of medical attendants working on-site during the accident was not reported, Dr Guskova stated that eight of them suffered from acute radiation syndrome. One, a young physician, later died. Some of the medical personnel in attendance at Hospital No. 6 in Moscow received 0.04- to 0.05-Gy (4- to 5-rad) total-body exposures and 0.35- to 0.4-Gy (35- to 40-rad) hand exposures during the first two weeks (Angelina Guskova, MD, oral communication, August 1986). These exposures primarily were incurred by handling patients with high burdens of internal contamination and/or contaminated damaged skin. Contamination control procedures were established in Hospital No. 6, and medical attendants were followed up with whole-body counting and urinalysis for possible exposure. These data have not yet been reported.

Before the Chernobyl accident, the focus on decontamination of patients was at the scene of the accident or at regional hospitals. It is now clear that contaminated patients may be transferred to definitive-care centers. Contamination control procedures must be developed in emergency departments and in hematologic and burn wards. These procedures can be an extension of those already present in medical centers for handling patients receiving radioisotopic treatment or brachytherapy (implant of a radioactive source, eg, cesium 137 or iridium 192, into tumor tissue).

THIRD PHASE OF MEDICAL RESPONSE

Within days, 450 medical "brigades" were organized and sent to Chernobyl from all parts of the Soviet Union. Each brigade consisted of a physician, a nurse, and a radiation technician. A total of 5960 medical personnel, including 1240 physicians, 920 nurses, and 2720 radiation technicians, were used.[11(p67)] The mission of these brigades was to attend to evacuees and site personnel who had not received doses of radiation sufficient to cause symptoms that led to hospitalization. These brigades were responsible for medical and radiologic examinations of the 135 000 evacuees and 100 000 children, many of whom were outside the evacuation zone. Though no children were reported to have experienced acute radiation syndrome, many were temporarily hospitalized for anxiety and other illnesses (Leonid Ilyin, MD, oral communication, August 1986). In the first weeks, over 100 000 thyroid scans and blood cell counts and more than 1000 whole-body counts were performed, the results of which have not yet been made available. The medical brigades were also responsible for supervising the detection of radioactive contamination, skin decontamination, and the distribution of potassium iodide to the off-site population.

The estimated individual exposures to the residents of Pripyat, who were advised six hours after the accident to stay indoors, averaged 0.033 Gy (3.3 rad). The comparable unsheltered dose from the time of the accident to evacuation was estimated to be 0.1 to 0.15 Gy (10 to 15 rad).[1(p40)] The average individual exposure for the 135 000 evacuees was estimated to be 0.12 Gy (12 rad).

POTASSIUM IODIDE

A large amount of potassium iodide 130-mg tablets was readily available for use during this accident. Potassium iodide was distributed to site personnel at 3 AM, 1½ hours after the accident, and to residents of Pripyat at 8 AM on April 26th, 6½ hours after the accident. It took much longer, from April 28th to the first days in May, to initiate and complete distribution of potassium iodide to the remaining population (90 000 persons) within 30 km. All recipients took one tablet of potassium iodide daily until May 6th. There were 2000 pregnant women among the 135 000 evacuees living within 30 km of the site and none were given potassium iodide. The average total-body dose for pregnant women was 0.43 Gy (43 rad) (Y. Lukanova, MD, unpublished teleconference, Sept 11, 1986). At this writing, 300 live births "with no obvious abnormalities" had been reported (*New York Times*, Feb 16, 1987, p10). At the April 1987 Southampton (England) Symposium of the British Institute of Radiology, it was reported that all newborns in the year since Chernobyl were normal.[12] The number of miscarriages and spontaneous or therapeutic abortions is not known. Also not known is the number of abortions induced for reasons of birth control.

In addition to reducing the iodine 131 uptake by the thyroid gland, the Soviet physicians felt that potassium iodide had a positive psychological effect on the population. They reported a number of minor side effects such as a metallic taste sensation and pharyngitis (Leonid Ilyin, MD, oral communication, August 1986). None of these side effects required medical attention. In northeastern Poland, however, severe iodine reactions occurred among 17 of the 10 000 000 potassium iodide recipients immediately after the first iodine 131 contamination of this area of Europe. These shocklike reactions required medical therapy (C. C. Lushbaugh, MD, oral communication, May 1987). The results of the thyroid iodine 131 uptake studies performed at two different times on the evacuees are seen in Table 3. The highest thyroid doses were reported in peasants who ignored the government warning and continued to drink milk from private cows. Because of the late administration of potassium iodide to the evacuees and the continuous releases of iodine 131 from the accident, the thyroid radiation exposure may be higher than was first calculated.

Potassium iodide is the recommended treatment for an accidental overexposure to iodine 131. If an exposure to iodine 131 will result in a thyroid dose of 0.1 to 0.3 Gy (10 to 30 rad) or greater, the National Council on Radiation Protection recommends that a 130-mg dose of potassium iodide be administered as soon as possible and repeated daily for seven to ten days.[13] If given within one hour, an effective thyroid block to iodine 131 uptake of 90% or more can be achieved; at four to five hours, the iodine 131 uptake can be decreased by 50%. Initiating the administration of potassium iodide after 12 hours will have very little effect.[14] Below an expected thyroid dose of 0.1 Gy (10 rad), adverse effects of the drug may outweigh radioiodine hazards.

Table 3.—Thyroid Iodine 131 Levels in Evacuees*

Thyroid Dose, μCi (MBq)	Persons Affected, %
April 26, 1987 (n = 171)	
<20 (<0.74)	87
<50 (<1.8)	94
100-150 (3.7-5.6)	1.5
>200 (>7.4)	0.5
May 6, 1986 (n = 104)	
<10 (<0.37)	90
10-20 (0.37-0.74)	5.2
20-50 (0.74-1.8)	4.8

*Measurements taken April 26 were from the population of Pripyat. Those taken May 6 were from the more distant populations. Data were presented by Angelina Guskova, MD, at the Vienna meeting.[1] (MBq indicates megabecquerel.)

OBSERVATIONS

The Chernobyl accident represented the first time a preplanned, organized emergency medical program responded to a nuclear accident involving mass radiation casualties. Though the Soviets probably never anticipated evaluating and treating hundreds of casualties, the program expanded to meet the need.

Three key organizational factors contributed to the success of the operation. First, and most important, the emergency medical response was coordinated and directed by a single highly qualified physician. With only one person responsible for coordination of medical activities at all locations, the opportunities for confusion and mismanagement were greatly minimized. Second, the design of the program provided that knowledgeable physicians would be immediately available both onsite and at regional hospitals to sort patients and provide early medical care. The presence of these physicians expedited early triage efforts and, in retrospect, saved lives. Since radiation injuries are seldom, if ever, immediately life threatening, patients with life-threatening nonradiation trauma received immediate medical attention. Those with less serious injuries and symptoms of acute radiation syndrome received attention as needed. The third factor was the availability and later arrival of a team of radiation medicine experts from Moscow to assist regional hospitals in further radiologic evaluation of people needing evacuation to predesignated definitive-care centers. That no more than 30 people died while under medical care is a tribute to Soviet planning and their expertise in radiation medicine.

The Chernobyl facility, when compared with Western commercial nuclear power plants, is unique. The large mass of combustible material, along with the lack of full containment, was the underlying basis for the severity and number of casualties. The burning graphite pile and the fires on the roofs of adjacent reactor buildings required heroic measures to control. Without these fires, it is unlikely that so many people would have been exposed to such high levels of radiation and that skin damage would have resulted from thermal and radiation effects.

The Soviet experience emphasizes the need to establish priorities in medical response. First, there is a need to train physicians living in the vicinity of nuclear facilities to provide and control the initial medical response for on-site victims, including triage and initial radiologic evaluation. This competence must be periodically maintained. Second, there must be available to these physicians "specialty" teams to assist them in secondary radiologic triage of patients. This triage, as at Chernobyl, should take place at local hospitals so that, except for those with major trauma, patients can be evacuated in an unhurried manner to definitive-care centers. These centers must be medically, radiologically, and psychologically prepared to receive and handle radiation injuries. Third, a number of definitive radiation medicine centers should be identified, such as centers for radioactively contaminated trauma and burned patients and centers for advanced hematologic care, a few of which need to include facilities for bone marrow transplantation. Having available a variety of centers alleviates the problem of overburdening any one center and would allow for flexibility and timeliness of evacuation. All medical facilities in the chain of evacuation, including definitive care centers, must be prepared to detect and control contamination. These medical facilities also must make provisions for monitoring attendant personnel for both internal and external radiation exposure over extended periods of time. A coordinated medical and health physics response should be set up ahead of time. Ad hoc arrangements after the fact can only lead to increased confusion and delay in proper patient care.

There has never been an accident in a US commercial nuclear power plant in which an employee developed symptoms as a result of exposure to ionizing radiation. The difference in the design of Western reactors (water-moderated vs the Soviet graphite-moderated reactor), the absence of a large amount of combustible material, and the development and exercise of emergency plans since the Three Mile Island accident minimize the risk of having a large number of employees with severe radiation injuries. However, an accident in a commercial nuclear reactor still has the potential to produce contaminated thermal burns, fewer but equally complex radiation injuries, and a larger number of asymptomatic exposures requiring medical and radiologic evaluation. Also, physicians can expect that an accident in the United States would result in intense media attention to patient care, a contingency not immediately faced by the Soviets.

PREPAREDNESS IN THE UNITED STATES

The Chernobyl accident has drawn the attention of the medical community to preparedness for nonmilitary nuclear accidents in the United States. The American Medical Association convened a committee on this subject and held a conference, "International Conference on Non-Military Radiation Emergencies," in November 1986 in Washington, DC. The proceedings of this conference will be available shortly. The Nuclear Regulatory Commission and the nuclear industry are also conducting a review of medical preparedness. In November 1986, the Federal Emergency Management Agency issued a guidance memorandum for medical preparedness at state and local levels.[15]

Presently, there are two major programs in this country with a committed 24-hour availability of emergency medical response for nuclear accidents. The Radiation Emergency Assistance Center/Training Site (REAC/TS), sponsored by the federal government, is located at Oak Ridge, Tenn; Radiation Management Consultants (RMC), developed by the nuclear industry, is located in Philadelphia. Radiation Management Consultants developed the regional approach to the management of radiation injuries.[16] Both RMC and the REAC/TS maintain "specialty teams" for dispatch to accident scenes or local hospitals to assist in radiologic evaluation and to identify patients for evacuation to definitive-care centers. Both have available an accident radiobioassay capability (including whole-body counting), radiation cytogenetic facilities, and external dosimetric evaluation to support clinicians in the definitive evaluation and treatment of radiation injuries. While the RMC program is primarily dedicated to accidents in commercial nuclear power plants, the major responsibility of the REAC/TS is to accidents in federal facilities and to the World Health Organization for accidents in the Western Hemisphere. Both organizations work closely to support each other should there be a shortage of resources.

Both organizations also conduct semi-

nars and training programs for medical personnel in the evaluation and treatment of radiation injuries. Radiation Management Consultants and the Radiation Emergency Medical Service Corporation, located in Albuquerque, also conduct semiannual on-site training and drills in the handling of radiation injuries at nuclear power plant sites, associated local ambulance organizations, and hospitals. Following are some of the medical centers that have both the clinical and radiologic capability to provide definitive evaluation and treatment of radiation injuries: The Hospital of the University of Pennsylvania, Philadelphia; Northwestern Memorial Hospital, Chicago; the University of Cincinnati Hospital Medical Center; and the Presbyterian University Hospital, Pittsburgh.[17]

To assess properly all of the medical-radiologic resources available and necessary to respond to nonmilitary nuclear accidents in the United States, a number of tasks need to be accomplished. Nuclear facilities where accidents could result in more than a few serious radiation injuries should be identified. This study should review potential accident scenarios, the number of people most likely to be involved, and the extent and severity of both radiation and nonradiation injuries.

A probability risk assessment would determine the expected frequency of such accidents. The availability of locally trained medical resources around each facility also should be identified. The study should also determine which other institutions and organizations provide emergency response and are prepared to serve as hematologic, burn, and bone marrow transplant centers for radiation injuries. The location of individual physicians and health physicists with experience in accident dosimetry and patient evaluation would also be a valuable part of this study.

In summary, it appears likely that there are ample medical and radiologic resources in the United States to cope with nonmilitary nuclear accidents. However, these resources are not always readily identifiable or organized to respond in a timely manner to provide the best patient care and alleviate the enormous anxiety that is sure to accompany a large accident. That is the lesson of Chernobyl!

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAR 3 1987

MEMORANDUM FOR: Charles E. Rossi, Assistant Director
Division of PWR Licensing-A

THRU: Victor Benaroya, Chief
Facilities Operations Branch

Scott Newberry, Section Leader
Facilities Operations Branch

FROM: Warren Lyon
Facilities Operations Branch

SUBJECT: STEAM GENERATOR TUBE RUPTURE DURING SEVERE ACCIDENTS AT SEABROOK STATION

REFERENCES:

1. Rossi, Charles E., "Steam Generator Tube Rupture During Severe Accidents at Seabrook Station - Draft Interim Report", NRC Memorandum for Vincent A. Noonan, Dec. 8, 1986.

2. Theofanous, T. G., "Review Comments on Seabrook Station Steam Generator Tube Response During Severe Accidents (a draft NUREG Report dated 12/15/86) and Related Sections of Technical Evaluation of the EPZ Sensitivity Study for Seabrook (a draft BNL Report dated 12/5/86), Dept. of Chem. and Nuc. Eng., Univ. of Calif., Jan. 12, 1987.

Plant Name:	Seabrook Station, Unit 1
Docket Number:	50-443
Resp. Directorate:	PWR Directorate #5
Project Manager:	Victor Nerses
Review Branch:	Facilities Operations Branch, DPL-A
Review Status:	Ongoing

We previously transmitted a draft assessment of Steam Generator Tube Rupture at Seabrook Station (Ref. 1). We have updated this document and have included changes identified by Theofanous (Ref. 2). The updated report is enclosed.

This report addresses the state of knowledge pertaining to Steam Generator Tube Rupture during postulated severe accidents, and the application of this knowledge to the Seabrook Station nuclear power plant.

There has been no attempt to comprehensively address all aspects of the subject, and many topics, such as operator actions, are merely identified as areas where further work is indicated.

Contact: W. Lyon
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ENCLOSURE

SEABROOK STATION STEAM GENERATOR TUBE RESPONSE DURING SEVERE ACCIDENTS

JANUARY 27, 1987

The major conclusions are as follows:

1. Study of SGTR due to severe accident conditions is difficult due to the complexity of the phenomena and the developmental nature of analysis techniques.

2. Further work is necessary to conclude that SGTR is unlikely under conditions associated with a severe accident.

3. SGTR due to severe accident conditions can be shown not to be a problem if the reactor coolant system is depressurized.

Warren

Warren Lyon
Facilities Operations Branch

Enclosure: As stated

cc: T. Novak
S. Long
F. Coffman
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V. Nerses
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Foreword

This report addresses the state of knowledge pertaining to Steam Generator Tube Rupture during postulated severe accidents (approach to core melt and core melt), and the application of this knowledge to the Seabrook Station nuclear power plant. This is an interim report, prepared with the assumption that the work and assessment will continue. The report does not cover all material received from Public Service of New Hampshire (PSNH) and its contractors, nor is it intended to provide a complete coverage of the issue. It does, however, identify a number of areas where work has been accomplished, it provides an assessment of that work, and it provides suggestions for future work which may be needed to resolve the issue. Actual resolution effort will depend upon addressing such issues as a pressurized Reactor Coolant System vs. one which has been depressurized.

Nomenclature

AC	Alternating current
ACRS	Advisory Committee on Reactor Safeguards
BNL	Brookhaven National Laboratory
EPRI	Electric Power Research Institute
ICC	Inadequate core cooling
KW	Kilowatt
LM	Larson Miller parameter
LOCA	Loss of coolant accident
MW	Megawatt
NRC	Nuclear Regulatory Commission
NSSS	Nuclear steam supply system
PDS	Plant damage state (See below)
PORV	Pressure operated relief valve
PRA	Probabilistic Risk Assessment
PSNH	Public Service of New Hampshire
RAI	Request for additional information
RCP	Reactor coolant pump
RCS	Reactor coolant system
RWST	Refueling water storage tank
SG	Steam generator
SGTR	Steam generator tube rupture
Staff	The NRC Staff

Plant damage states are used to classify conditions as follows:

1 Early core melt, low RCS pressure at time of reactor vessel failure, RWST injection not initiated

2 Early core melt, low RCS pressure at time of reactor vessel failure, RWST injection initiated

3 Early core melt, high RCS pressure at time of reactor vessel failure, RWST injection not initiated

4 Early core melt, high RCS pressure at time of reactor vessel failure, RWST injection initiated

5 Late core melt, low RCS pressure at time of reactor vessel failure, RWST injection not initiated

6 Late core melt, low RCS pressure at time of reactor vessel failure, RWST injection initiated

7 Late core melt, high RCS pressure at time of reactor vessel failure, RWST injection not initiated

8 Late core melt, high RCS pressure at time of reactor vessel failure, RWST injection initiated

9 Core melt with non-isolated SGTR

A Containment intact at start of core melt, containment heat and fission product removal available

B Containment intact at start of core melt, containment heat removal only available

C Containment intact at start of core melt, containment fission product removal only available

D Containment intact at start of core melt, none of the containment functions available

E Containment not intact at start of core melt, activity release filtered

F Containment not intact at start of core melt, containment opening larger than three inch diameter

FP Containment not intact at start of core melt, containment opening smaller than three inch diameter

FA Aircraft crash

1. OVERVIEW AND SUMMARY

The Public Service of New Hampshire (PSNH) has presented information to show that the Seabrook Station containment is one of the strongest of any nuclear power plant. It also contains one of the largest volumes. This combination leads to a conclusion that the containment has the capability to either significantly delay or prevent the release of large quantities of radioactive material during and following a severe (core damage or core melt) accident. Based on this premise, any significant risk associated with Seabrook Station will likely be found in accidents which bypass containment.

Recognizing this, the Staff and PSNH have explored containment bypass possibilities. One possibility, the topic of this report, and a potential issue that has been under investigation by industry and the Staff for several years, is the loss of steam generator tube integrity due to generation of high temperatures at high pressure during a core melt accident. The potential concern involves movement of high temperature fluid from the region of the melting reactor core into the steam generator tubes, with a resultant over-heating of the tubes which leads to their rupture. High pressure fluid containing radioactive material from the melting core would thereby be released to the secondary side of the steam generators, from where it could be released to the environment via the steam generator relief valves, thus bypassing containment.

For steam generator tube rupture (SGTR) to be a concern as addressed here, one must have a core damage (or melt) condition in progress with no water on the steam generator secondary side. The principal contributor to this condition is estimated to be a loss of all AC power concurrent with a loss of all turbine driven feedwater to the steam generators. PSNH has investigated the possibility of encountering conditions which can contribute to SGTR and has determined the likelihood to be less than 4×10^{-5} per reactor year. This is sufficiently high, and the potential consequences of SGTR under severe accident conditions are sufficiently great, that further investigation has been necessary. This investigation is ongoing. This report provides an interim assessment of the status of the investigation, as well as a projection of expected results.

Study of SGTR due to severe accident conditions is difficult. The phenomena are complex, and most analysis techniques used to investigate nuclear power plant behavior have utilized assumptions which are not applicable here. The principal complication is the multidimensional character of fluid behavior in the reactor coolant system. Suitable computer programs are just beginning to become available. Suitable experimental information is just being developed. Hence, pioneering work, such as provided by PSNH in investigation of this issue, can be expected to have weaknesses as well as strengths. We have found this expectation to be true.

The work reported by PSNH and its contractors is highly informative and addresses most aspects of the SGTR issue. It is based upon knowledge of what takes place within the Nuclear Steam Supply System (NSSS), upon a major computer program that is under development and is being verified (MAAP), and upon information derived from an experimental program at Westinghouse. The following is a summary of the reported information and our assessment:

1. <u>Mathematical modeling.</u> Expected phenomena, experimental information pertinent to the phenomena, and modeling assumptions have been addressed for each of the major components of the NSSS which are affected. Multidimensional fluid flow and energy transport have been established as dominant over most of the conditions of interest. We consider this area to be in a preliminary stage of development, and there are some potential difficulties, which include:

    a. Certain modeling assumptions are overly optimistic. An example is the assumption of complete mixing in the steam generator inlet plenum which tends to reduce the temperature of fluid entering the steam generator tubes. This assumption is not supported by the available experimental evidence, and the effects of the assumption are not balanced by identifiable pessimistic assumptions elsewhere in the analysis.

    b. Experimental evidence is preliminary. The experimental facility at Westinghouse is providing information pertinent to this issue. However, testing has been limited to conditions which are only

roughly scaled to NSSS representation. This is due to a logical progression in the test planning and facility development. Data from apparently well scaled test conditions are just becoming available. No other test facility addresses certain aspects of this issue.

c. The computer program used as the basis for much of the work has not been verified, nor is documentation available. We understand a verification program and an effort to provide documentation are underway. (PSNH contractors have offered to discuss this information with us. Our review has not progressed to the stage where we can make use of this offer.) Although the phenomena we understand to be modeled by the code appear adequate for the purposes needed here, and the code results appear reasonable subject to our concerns as expressed elsewhere in this report, this is not sufficient information to accept the analysis results.

2. Seabrook Station Representation. The basic analyses and sensitivity studies have been based upon a plant configuration in which the NSSS state is assumed. Most of the assumed state conditions are reasonable. There are exceptions. For example, the steam generator secondary side is assumed to be at a pressure corresponding to secondary side relief valve settings, and creep rupture of tubes is reported for this state. The resulting conclusions are similarly based upon this state. We believe there is sufficient likelihood the secondary side will be depressurized that this case should be considered. Depressurization would roughly double tube stress since the secondary side pressure would be decreased from roughly 1100 psi to atmospheric pressure while the RCS pressure remained at approximately 2300 psi.

3. Sensitivity Studies. PSNH and its contractors have performed a wide ranging sensitivity study as part of an assessment of the impact of various modeling assumptions and the state of the plant. Although this yields valuable information and insight, sensitivity studies should be

approached with caution. They are only as good as the basic modeling. The impact of our difficulty with assumptions such as the behavior of the steam generator inlet plenum is not addressed in the sensitivity study, and could impact the results and conclusions.

4. Operator Actions. Plant response can be drastically altered by operator actions during a severe accident. SGTR is no exception. A number of operator responses have been discussed with PSNH. Although many of these were postulated actions, significant information has been developed from these postulations. Recognition that operator actions could depressurize the steam generator secondary side is one item raised during the review. Depressurization of the reactor coolant system via the pressurizer Pressure Operated Relief Valve (PORV) to avoid the SGTR problem is another.

We find that the topic of SGTR is in a developing state, with knowledge being rapidly accumulated. Further work is necessary to conclude that SGTR is unlikely under all conditions associated with a severe accident.

Existing knowledge can be used to support a conclusion that SGTR is not a problem if the RCS is depressurized. Consequently, reasonable assurance that progressions toward core melt would not occur at high RCS pressure, coupled with supporting evidence in regard to steam generator tube response, would alleviate our concern regarding SGTR under severe accident conditions. We have not conducted an evaluation of the trade-offs associated with such an approach, nor have we been provided with information that would either support or negate RCS depressurization under severe accident conditions. We have not provided a recommendation regarding whether RCS depressurization is attractive when all pertinent factors are considered.*

Our judgement is that a carefully conducted thorough evaluation on the part of PSNH can establish that the likelihood that a SGTR will result due to overheating during severe accidents which initiate from power operation is sufficiently small that the risk associated with this event can be shown to be negligible. Our judgement is preliminary and has not been substantiated.

* Theofanous (Ref. 22) believes depressurization should be accomplished, and does not foresee any significant reasons why this should not be done.

Determination of the correctness of a judgement regarding SGTR under severe accident conditions originating from power operation with the RCS at high pressure can be based upon a combination of analytic and experimental investigations. The ongoing test at Westinghouse in which reasonably close similitude is claimed between the test facility and appropriate parts of a Westinghouse four loop NSSS will provide key data which can be applied to assist in the development and confirmation of analysis techniques. Use of selected test data from other facilities and further examination of the analysis techniques, coupled with necessary changes when they are uncovered, should provide sufficient confirmation that reasonable reliance can be placed upon accident analyses pertinent to this issue. Suitable analyses can then provide a sufficient foundation to resolve this issue.** Such a program will represent a formidable undertaking.

** Theofanous (Ref. 22) states this judgement to be "... an overly optimistic and inappropriate judgement for the Regulatory to make at this time". He continues with "...the procedure outlined to 'substantiate this judgement' is unrealistic and incomplete". Although we continue to believe the issue can eventually be established to not contribute significantly to risk, we certainly agree with Theofanous' assessment that such a determination will not be easy. Further, our "judgement" is preliminary and unsubstantiated, and is not to be used as the basis for any regulatory findings until reasonably established to be incorrect or correct.

2. INTRODUCTION

The Public Service of New Hampshire (PSNH) reporting of Seabrook response to accident conditions in References 1 - 4 represents one of the most comprehensive investigations of nuclear power plant accidents in a specific plant that we have encountered. Some accidents which have a significant impact upon risk are treated more comprehensively than previously reported by any investigator. For example, References 3 and 4 describe an investigation of LOCA outside of containment that is more comprehensive than any we have reviewed. Many of the commonly used conservatisms, which distort the perception of accident impact, have been removed. What results is a serious attempt to better represent plant response to severe accident conditions, with particular attention to items which have previously been identified as having a serious impact upon risk.

PSNH has presented information to show that Seabrook Station has one of the strongest containments of any nuclear power plant. It is also one of the largest with respect to containment volume. The combination of large volume and strength leads PSNH to a conclusion that the containment can mitigate virtually every severe accident and, at the worst, can significantly delay release of meaningful quantities of radioactive material during anc following core melt accidents. Most core melt accidents can be contained within the Seabrook Station containment, and, if this is accomplished, little radioactive material will escape. The full mitigative capability of the Seabrook containment will be realized if there are no "holes" in the containment. Such holes can exist if any of the following occur:

1. Containment is not properly closed (isolated), such as can occur if containment ventilation is not properly closed upon receipt of a containment isolation signal,

2. A failure occurs which allows the containment atmosphere to escape, such as failure of a containment penetration due to a combination of high pressure and high temperature, or

3. A failure occurs which allows material to move directly from the Nuclear Steam Supply System (NSSS), principally the Reactor Coolant System (RCS), to the environment, such as occurs with the traditional "Event V" (Ref. 5), with LOCA outside containment leading to core melt and the release of radioactive material via the LOCA flow pathway.

Clearly, if PSNH conclusions regarding containment strength are verified, there will be little risk associated with accidents at Seabrook Station unless containment is bypassed. Therefore, core damage accidents with containment bypass deserve careful attention. PSNH has reported studying some bypass accidents in detail (Refs. 3, 4, 12, 17, and 18). Such studies have led them to conclude that certain bypass accidents at Seabrook, such as LOCA outside containment, engender significantly less risk than previously believed. Other bypass accidents have only recently been identified and accident investigation is not complete.

One potential area for bypass, as identified above, involves paths between the RCS and the environment. Certain phenomena can potentially lead to such paths. These involve multidimensional fluid behavior and fission product transport within the RCS during the approach to core melt and during the core melt process. Consideration of these phenomena has a significant impact upon RCS response, including potentially the location of RCS failure. There are many possible implications, including the possibility that the impact of RCS failure on containment may have been overestimated in past analyses. The implication of interest here is that failure to accurately model RCS fluid and fission product heating behavior might result in an RCS failure which bypasses containment. The only area discovered where this is of immediate concern involves the Steam Generator (SG) tubes. If these fail during a core melt accident while the RCS is at high pressure, there is a high potential of a major release via the SG relief valves or the SG Pressure Operated Relief Valves (PORVs), which vent directly to the environment; or via a rupture in a steam line outside containment.

The general concern addressed in this report is the rupture of multiple SG tubes in response to high temperature, which in turn is a result of core

uncovery. This accident sequence should be of concern any time there is a core melt with the RCS at high pressure in combination with no water in the SG secondary sides. These conditions lead to a potential for natural circulation transport phenomena to significantly heat the tubes prior to breach of the reactor vessel. If this occurs, the resulting loss of tube strength could lead to tube rupture. If tube rupture occurs, and any of the secondary side valves are open, the secondary side is breached outside containment. Alternatively, if the RCS pressure is above the SG relief valve setpoints, containment is similarly bypassed. There is no substantiation which establishes that these valves will close after being exposed to such an environment, nor has it been established that other secondary side failures will not occur. This area has not been adequately investigated, and is not recognized as a release path in the early Pickard, Lowe and Garrick work on risk investigation at Seabrook Station (Refs. 1-4), nor is it addressed in any of the other PRAs we have received. It has been addressed in more recent work (Refs. 12, 17, and 18).

The concern was expressed as the rupture of multiple steam generator tubes. We do not believe single tube ruptures will occur under the severe accident conditions of interest. The reason for this is that if one tube ruptures, or even begins to leak significantly, this will induce flow of hot RCS fluid toward the leak. Therefore, the location of tube rupture will probably quickly become hotter. If high temperature is what led to the break, a higher temperature can only make it worse. Tubes in the vicinity of the break will be exposed to the high velocity break flow, in additional to high temperature, weakening them and, we believe, quickly leading to their failure. We believe this cascading effect would rapidly propagate to multiple tube rupture, stopping only when sufficient RCS depressurization has occurred that tubes are no longer stressed by a significant pressure differential across their walls.*

*This belief is based in part on the assumption that SG tube degradation has been controlled and there are no "outliers" which fail significantly sooner than other tubes due to existing tube imperfections.

Although this report is limited to SG tube rupture, there are other SG components which separate RCS fluid from the SG secondary side. These components, such as the SG tube sheet, must be investigated to achieve completeness in the investigation of containment bypass via the steam generator.

An initial consideration in investigation of the SG tube rupture issue is "What is the likelihood of attaining conditions where SG tube response could be of concern?" Principally, the conditions are loss of all SG feedwater with a simultaneous loss of RCS makeup capability; conditions which result, for example, from a loss of all AC electrical power with the simultaneous loss of the turbine driven auxiliary feedwater pump. PSNH estimated this condition to have a mean annual frequency of less than 4.5×10^{-5} per reactor year ** (Ref. 17). A value of this magnitude is sufficiently high that tube response must be considered.

** We have not fully investigated this value or its uncertainty and consequently are not verifying it as "correct" via its usage here. We do believe it is of a reasonable magnitude, and as such, that further work on SG tube rupture is indicated.

3. STEAM GENERATOR TUBE RUPTURE (SGTR) UNDER SEVERE ACCIDENT CONDITIONS

3.1. Description of Phenomena and Potential Concern.

The RCS is generally modeled with a one dimensional representation of fluid flow, and in some cases with parallel one dimensional modeling in regions such as the reactor vessel. This has been particularly true for PRAs, where to our knowledge, all have been based upon computer code analyses which incorporated single dimensional representations of fluid behavior within the RCS. Additionally, movement of the source of heat due to fission product migration is seldom modeled.

The possibility of RCS behavior being different from what is generally represented during severe accidents has been recognized for some time. Winters (Ref. 6) identified aspects of the problem in 1982. Denny identified potentially important aspects of natural circulation, and Denny and Sehgal (Ref. 7) provided preliminary multidimensional analysis results in 1983. The topic was discussed by an NRC containment response working group and with the ACRS (Refs. 19 and 20), it was the subject of an NRC/Industry meeting (Ref. 8) and a formal request for work within NRC (Ref. 9), and SGTR possibilities were identified (Ref. 21) in 1984. Potential impact upon SGTR was estimated on a preliminary basis (Ref. 10), and experimental data were presented from an ongoing series of tests (Ref. 11), in 1985. Numerous analysis results have been published since the early publications of Denny and Sehgal which represent work sponsored by both industry and the NRC. However, there is no published analysis of overall NSSS response to a broad range of severe accident conditions which includes these phenomena, and which is based upon accident analysis methods which have been subjected to broad peer review and acceptance. This introduces a difficulty into review of SGTR during severe accidents with respect to the impact upon the Seabrook Station risk evaluation. As will be seen, sufficient work has been accomplished that what appear to be reasonable conclusions can be formulated, although confirmation will require additional effort. As will further be seen, there appear to be operational methods which can negate the problem, although the impact on other aspects of plant operation has not been evaluated.

The potential misrepresentation of system response of concern here stems from the fluid flow behavior inherent in one dimensional modeling as utilized by most accident analysis codes. Such modeling typically represents flow through the reactor core as determined by the water boiloff rate from the lower core or lower plenum. This rate becomes small as the water level approaches the bottom of the core. Typical calculations (see historical references which were previously discussed) indicate that the flow rate due to natural convection which occurs in a multidimensional manner is of the order of ten or more times that of the flow due to boiloff. Hence, the calculations are typically based on a minor contributor to flow, and the major contributor is neglected.

The modeling difficulty also applies to upper plenum behavior. One dimensional modeling of any fluid (liquid, vapor, or gas) that passes through the core is typically assumed to flow through the upper plenum and out the hot leg.

This modeling is incorrect under severe accident conditions where a major portion of the core has been uncovered or the core is being vapor or gas cooled since strong recirculation patterns will develop which thermally link the core and upper plenum. At pressures in the range of 2250 psi, the linkage is strong, and some of the upper plenum component temperatures can be expected to closely follow core temperature during the early stages of the approach to core melt. The strength of the linkage diminishes with decreasing pressure. Information also exists which illustrates a decrease in linkage with increasing hydrogen concentration and core damage (although initial production of hydrogen may enhance circulation due to the buoyant gas "pushing" its way toward upper regions of the reactor vessel).

Correct consideration of the hot leg and steam generator behavior leads to calculation of significantly different behavior when contrasted to one dimensional modeling. Hot fluid, at a temperature far greater than predicted via a one dimensional model, will enter the upper portion of the hot legs from the reactor vessel, and flow toward the inlet plenum of the steam generators. Displaced colder fluid will return to the reactor vessel upper plenum along the bottom of the hot legs. Circulatory patterns will become established in the steam generator inlet plena in which some of the hot incoming fluid is mixed

with plenum fluid. Fluid from the steam generator inlet plena will flow into some of the steam generator tubes in the nominal forward direction, displacing fluid in the steam generator outlet plena. This displaced fluid will flow through other tubes in a nominal reverse direction, reentering the steam generator inlet plena. (All of these flows have been observed experimentally as described in References 11, 13, and 14). This mechanism has the potential to transport hot fluid from the reactor vessel into the steam generator tubes during core heatup and melt, with the result of creating the potential of overheating the tubes if there is no water on the steam generator secondary side.

There are other possibilities which could challenge tube integrity as well. For example, RCP seal LOCA or a small RCS cold leg break introduce a low pressure region between two regions where a liquid seal or plug may exist -the crossover pipe between the RCP and the SG, and the lower reactor vessel. Under approach to core melt conditions, one path for flow is through the SG tubes, through the crossover pipe seal, and out the break. (Note this does not remove the seal - the steam simply bubbles through it). This flow path of hot steam through SG tubes and the associated thermal impact on the tubes must be considered. Another tube challenge can result due to emergency procedures. Many plant Inadequate Core Cooling (ICC) emergency procedures specify RCP operation if conditions exist which indicate an approach to core melt, and alternate mitigative measures have failed. Such a step could circulate hot fluid through the RCS, including the tubes. Although this may slightly extend the time to core melt, it may be an unattractive approach if it also introduces a high likelihood of loss of tube integrity. To our knowledge, these contrasting responses and the impact upon risk have not been studied. (Note the likelihood of encountering the emergency procedures problem situation is small, but it does exist.)

A final phenomenon that has received inadequate attention during conditions leading to core melt is fission product movement. Typical one dimension accident code calculations take such movement into account from the viewpoint of radiological hazard, but do not include the influence upon heat generation. Approximately a quarter of the heat producing radioisotopes probably has left

the core under the conditions of interest, and substantial deposits can be expected in the upper plenum structure. This could have a significant influence upon thermal response, particularly if some of this material leaves the reactor vessel and enters the hot legs.

As will be seen in the following sections, PSNH has addressed many of these issues in the most comprehensive study of this problem that we have encountered.

3.2. Seabrook Station Steam Generator Integrity

3.2.1 Issues Addressed By PSNH

The PSNH has addressed many of the issues applicable to SG tube response to severe accident conditions (Refs. 12, 17, and 18). Analysis results were summarized which were intended to determine the thermal response of SG tubes under severe accident conditions. Basic analysis assumptions pertinent to the state of the plant were:

1. The steam generators must be dry to experience a significant thermal transient since, if the SG secondary side contains water, the tubes cannot overheat.

2. Station blackout conditions (Loss of all AC power) exist.

Analyses were conducted for the following:

1. Station blackout without operator actions or RCP seal LOCA

2. Station blackout with a 50 gpm RCP LOCA (each RCP) and no operator actions

3. Station blackout with operator actions

4. Uncertainty evaluation

Possible operator actions considered included:

1. Start steam turbine driven auxiliary feed water flow
2. Restore emergency AC power (diesels and/or switchgear)
3. Shed nonessential loads
4. Open RCS POPVs when core exit temperatures exceed 1200^{o}F.

A number of other operator actions one might expect were discussed during a meeting with the PSNH at BNL on October 17, 1986, including:

5. SG blowdown and depressurization to enable filling the SGs by the condensate booster pumps or from fire water systems. (There are two diesel driven pumps and one electrically driven pump at Seabrook Station. The ability to use these for injection into the SGs has not been confirmed.)

6. RCP operation, a step that is not possible unless off site electrical power has been restored. (PSNH felt the likelihood was sufficiently low that there would be negligible effect on risk.)

3.2.2 Likelihood of Conditions Leading to Tube Failure

PSNH addressed the question of conditions necessary for SGTR in the response to the Staff Request for Additional Information (RAI) 47 (Ref. 17). In this response, PSNH stated the risk to be small for the following reasons:

1. The frequency of high pressure core melt with dry steam generators is very small.

2. Given the postulated occurrence of a high pressure core melt with dry steam generators, creep rupture of the SG tubes is not a credible failure mode.

3. A large number of tubes must fail to produce an early large containment bypass.

4. All three of the following must occur in order for there to be a containment bypass:

   a. Failure to recover water to the SG
   b. Failure to depressurize the RCS
   c. SG tube creep failure

3.2.3 PORV Considerations

PORV operation as identified in item 4, above, is not specifically contained in Seabrook Station emergency procedures, but is believed by PSNH to be a logical operator response as an attempt to depressurize and obtain water from the accumulators. (Operator monitoring of the temperatures is specifically identified in the procedures for loss of all AC power conditions.) In addition to potential core cooling via the accumulator water, opening the PORVs is claimed to have the following effects:

1. It reduces stresses in all primary system components

2. PORV flow overrides natural circulation such that high fluid temperatures are not attained in the SGs, including the tubes.

In response to a staff question, PSNH indicated that the likelihood of being able to open the PORVs under loss of AC and ICC conditions was high. They also indicated that one PORV was sufficient since its "worth" is about 50 MW of energy removal in the form of steam, and have presented blowdown rate information in Reference 18. (Note Seabrook is equipped with two PORVs.)

Although we consider the EPRI funded Westinghouse tests pertinent to this issue to be somewhat preliminary with respect to scaling to NSSS conditions, some interesting effects have been observed that are worth noting which pertain to PORV operation. These include:

1. Natural circulation flow restores itself readily to the pre-opening condition in the hot legs, core, and communication paths between the upper plenum and the upper head following PORV closure.

2. Heat transfer in steam generators between the primary and secondary side fluids increases 50% to 75% with periodic venting.

3. The core is little affected except for the boundary with the hot leg that connects to the pressurizer surge line.

Item 2 is of particular interest since it carries an implication that flow in the steam generator tubes is enhanced by PORV operation (as well as by opening and closing of RCS safety valves). Hence, if one visualizes opening and closing a pressurizer PORV when degraded conditions are well established with the steam generator secondary side depressurized, there may be a tendency to enhance flow of hot RCS fluid through the tubes, with the potential of causing tube rupture.

3.2.4 Loop Seals

Loss of RCS inventory under natural circulation conditions (RCPs not running) is expected to leave the RCS in a condition where water is trapped at low elevations. According to a number of preliminary analyses, such loop water seals or plugs exist at the cross over leg between the SG exit and the RCP inlet, and in the lower region of the reactor pressure vessel. The absence of these water seals could significantly change circulatory conditions during ICC conditions, with the potential for changing SG tube response. Although we expect a careful examination of behavior in the Seabrook RCS would establish that the seals will remain under most boil down conditions, this expectation needs to be substantiated by suitable analyses which address the range of conditions which can exist during severe accidents.

Complete loss of the RCS liquid inventory with the RCPs running, followed by loss of the RCPs, could result in a homogeneous fluid condition in the RCS. Under this condition, fluid heated in the core would flow into the

upper plenum, through the hot legs, the steam generators, the RCPs, and back into the reactor vessel and the core via the cold legs. Although multidimensional fluid flow conditions probably exist in the reactor vessel after RCPs are lost, one may estimate that thermal response is still reasonably realistic if modeling is restricted to one dimension provided the natural convection flow rates are high. For this case, existing analysis codes could be applied to roughly estimate steam generator tube response. If the response was not clear, then multidimensional analyses could be applied to estimate the influence. In such a case, uncertainty in the multidimensional analyses might not be of as great a concern as for the situation of multidimensional behavior dominating system response. However, nonexistence of the loop seal due to continuous RCP operation is an unlikely situation since the majority of conditions during which steam generator tube integrity is of concern will involve loss of off site AC power, and RCPs will be unavailable. To our knowledge, a complete, accurate, analysis of a four loop Westinghouse NSSS has not been performed for these conditions. In additional to an analysis approach, closure of consideration of this aspect of SG tube behavior could be obtained if the probability of occurrence of the RCS homogeneous fluid condition was established as negligibly small in contrast to other situations where SG tubes were shown to lose integrity, or if the risk associated with the condition was established as negligible when compared to other Seabrook Station risks.

A second situation involving free circulation in the RCS might be obtained if one considers the RCPs as being restarted in response to high core temperatures, as prescribed in the emergency procedures. For this case, sufficient head might be developed to clear the loop seals of water, and rehomogenize the RCS fluid, thereby generating the condition described in the previous paragraph. To our knowledge, rehomogenization under these conditions has not been established to occur at Seabrook. Insofar as SGTR at Seabrook is concerned, the issue can be dealt with as outlined in the previous paragraph.

A third situation of removal of loop seals also potentially exists during boil down of the RCS inventory. One may postulate that the ICC condition occurs with the loop seals in place, and that some other mechanism causes their disruption. This could occur if a sufficient pressure difference occurred across the seals that they were forced out of the low regions or if superheated steam passes through the water, thus evaporating it. Several analyses have been conducted which include consideration of some of this behavior, and none showed loss of the seals. To our knowledge, these analyses have not carefully considered the evaporation question or the impact of a sudden pressure surge due to core slump into water in the lower plenum. One would expect that consideration of this condition could be closed if analyses applicable to Seabrook could reasonably establish that the seals remain.

Another condition can be visualized if one considers a LOCA to have occurred in the RCS. For example, a small cold leg LOCA (or an RCP seal LOCA) could be located between the two natural seal regions of the crossover leg and the reactor vessel lower plenum. Removal of RCS mass might occur under conditions such that the seal water was evaporated from the crossover leg due to forcing superheated steam through the seal water. An important aspect of seal behavior to consider here is that one does not have to empty the crossover leg of water to pass steam through the SG tubes. It is sufficient to bubble steam through the seal water. Elimination of consideration of this effect with respect to impact upon risk could be considered on the basis of a thermal-hydraulic investigation of RCS behavior, establishing that the potential impact on risk of the behavior is negligible in comparison to other established risk contributors, or both.

3.2.5 PSNH Modeling Considerations

The PSNH has reported application of the MAAP 3.0 code to investigation of natural circulation flow in Seabrook (Refs. 12 and 17). This code treats the major phenomena, including approximations of multidimensional flow and fission product (heating) movement, and is applied to the regions of the RCS which are affected by the SGTR issue.

Quasi-steady momentum balances and continuity equations are used to represent natural circulation flow, and the steam generator inlet plenum behavior is represented by quasi-steady mixing models. The modeling represents gas and wall temperatures using conventional lumped parameter models, with 15 gas control volumes and 17 two dimensional heat sinks. (Several volumes are subdivided into further volumes for some types of calculations. The core, for example, contains 70 nodes which comprise the core volume node.) The control volumes are based upon approximations of the flow patterns which were seen in the Westinghouse experiments on a scaled NSSS (Refs. 11, 13, and 14). This basis for definition of control volumes means that deviations from the assumed flow pattern and flow instabilities may not be represented in the model. Experimental evidence shows that there are asymmetric flow patterns, for example, which are not modeled, and which could lead to tube heating conditions which would not be calculated. Further, although instabilities have not been experimentally observed at the Westinghouse test facility, one must accept this evidence with care since testing with fluid conditions which closely simulate those expected in an NSSS are just being initiated.

Use of the lumped parameter model requires further discussion. Unlike computer codes such as COMMIX, which can determine flow patterns within certain bounds provided the configuration is properly modeled, a lumped parameter model is based more strongly upon a presupposed flow behavior. Although such representation can be valuable and accurate under certain conditions, such assumed behavior must be verified before it can be accepted. The preliminary Westinghouse experiments, as discussed briefly in the next section of this report, and some COMMIX and MELPROG calculations (Refs. 15 and 16), represent steps in this direction, but further evidence is necessary before we can accept the assumption as verified. (The experiments are somewhat preliminary, and the COMMIX and MELPROG calculations have not, to our knowledge, been carefully checked against experimental evidence.) We further note that, to our knowledge, there has

been no independent study of the version of the MAAP code used for the analyses. At a minimum, we believe a reasonable knowledge of code modeling and logic, in addition to a verification program, are necessary for acceptance of the calculated results. (We note that EPRI has a MAAP verification program underway.)

One aspect of the modeling appears worthy of further consideration. The steam generator inlet and outlet plena are assumed to be completely mixed in the PSNH studies being reviewed here, and they are represented by single nodes with uniform properties. The Westinghouse facility test data indicate a partially stratified, partially mixed SG inlet plenum (Ref. 14), and modeling for the test facility is based upon a quasi-steady state model in which partial mixing is assumed at various (limited) locations between streams of different origins. Reference 14 describes the situation as follows:

> "The flow in from the hot leg rises rapidly in a plume in the inlet plenum and induces mixing. Some of the cold return flow from the tube bundle does avoid mixing, particularly near the divider which is furthest from the hot leg. Much of the cold return tubes' flow plunges through the hotter stratified fluid layer that spreads across the bottom of the tube sheet. The mixing flows could be observed from dye injection and from observation of light through the density gradients that resulted. Temperature measurements in the inlet plenum are indicative of mixing. The tubes carrying hot fluid from the inlet plenum were generally concentrated in the area above the hot leg entrance and scattered in the regions further away. Cold return tubes were also scattered and were found in the area above the hot leg inlet also."

Test facility modeling of the phenomena uses a six equation approximation which contains an experimentally determined mixing parameter.
We believe the assumption of complete mixing used for the PSNH investigations will reduce SG tube temperatures when contrasted to the experimentally identified situation. This modeling and its implications need

further consideration. (This comment is repeated a number of times in the discussion of calculated NSSS response in the following sections of this report.)

3.2.6 Comparisons of Calculations to Experimental Data

Several comparisons between MAAP code calculations and experimental data have been briefly described by PSNH and its contractors to the BNL and NRC staffs (Refs. 12, 17, and 18). These are discussed below.

1. Core and upper plenum flow rates. The following comparison of experimental and calculated values was presented:

Test Condition	Experimental Flow Rate	Calculated Flow Rate
28 KW Water Test	0.54	0.50
0.9 KW SF_6 Test	0.016	0.017

2. Hot leg and steam generator natural circulation. Comparison of several parameters was provided:

Item	Experimental Value	Calculated Values for Indicated Number of Steam Generator Tubes Carrying Flow in the Out Direction		
		6	12	24
Heat Transfer Rate, KW	2.43	2.0	2.6	2.9
Entering Fluid, ^{o}C	30	30.7	29.2	28.4
Exiting Fluid, ^{o}C	19	24.2	21.7	18.8
Coolant, ^{o}C	10 - 11	9.4	11.2	12.8

where the entering fluid is flowing into the steam generator inlet plenum from the upper portion of the simulated hot leg, and the exiting fluid is flowing from the lower portion of the steam generator inlet plenum back

toward the simulated reactor vessel along the bottom of the hot leg. The coolant temperature is that of the water leaving the secondary side of the simulated steam generator, and thus, can be related to the heat transfer rate from the primary to the secondary sides.

These results are clearly promising. Continuation of the comparisons with a wide range of experimental conditions in the same test facility, and with no changes in the modeling except for the change of experimental conditions and fluid properties, would be helpful in code verification. Extension of the same modeling approach to other experimental data (such as flow in ducts and components) would provide further confirmation. Completion of confirmation of modeling adequacy could typically include comparisons of existing data obtained in large facilities, selected contrasting of alternate calculational methods to portions of the code under consideration here, and establishment that scaling is adequately represented by the code.

3.2.7 Calculated Seabrook Thermal Response to Severe Accidents

Calculated behavior to selected accident conditions has been summarized by PSNH. Principal results and our comments are as follows:

1. Peak Steam Generator Temperature for Loss of AC Power and Loss of Feed Water Flow. The following temperatures and flow rates were calculated at the indicated condition:

Location	Temperature, ^{o}K	Flow Rate, kg/sec
Core (Peak)	1800	18 (recirculating between
Upper Plenum	1160	upper plenum and core)
Hot Leg	760 (wall)	2.4 (countercurrent)
SG Inlet Plenum	850	-
SG Tube	700 (wall maximum)	3.3 (total in each direction)
SG Outlet Plenum	640	-

PSNH indicated that the hottest core node would melt at about 30 seconds from the time of these values, and that the generated hydrogen and blockage due to relocated core material would cause natural circulation between the core and the upper plenum to almost stop. At this point, the upper plenum would begin to cool due to energy transfer to the hot legs.

Plys (Ref. 18) presents additional information which shows temperatures continue to increase after vessel blowdown, with the peak upper plenum temperature exceeding 1200^{o}K for a short time. The tube temperature continues to increase for the time of the calculation (20,000 sec, with vessel rupture at 11,600 sec), reaching a maximum of about 1020^{o}K. We would be interested in seeing plots of other parameters over the span of the calculations, including the hot leg and SG plena temperatures, to better understand the interactions and modeling.

In response to a question, PSNH indicated they had not performed a detailed analysis of reactor vessel hot leg nozzle thermal behavior, but felt a temperature of the order of 1000^{o}K was necessary to cause failure. Discussion also identified that there was significant steam circulatory flow in the secondary side of the steam generator tubes, and that this steam, which was at a pressure corresponding to the steam generator safety valve settings, represented a significant heat sink. Further, it was an effective medium for transferring heat from hot tubes to colder tubes, thus tending to reduce the maximum tube temperature. This raises a question of what results would be obtained if the steam generators were depressurized to atmospheric pressure, thus maximizing pressure differential across the tubes and simultaneously removing a heat sink which could influence temperatures throughout the NSSS. (A sensitivity analysis was conducted in which this was one of the parameters.)

Information presented in Reference 12 and the above summary table shows fluid flow rates in the hot leg of roughly 2 kg/sec as contrasted with a rate above 3 kg/sec in the SG tubes for the time after effective boiloff of water from the core until melt through of the reactor vessel. Cooling

via steam contained in the SG secondary side is thus an effective medium for cooling the SG inlet plenum. The total mixing assumption pertinent to fluid in the plenum is, in turn, effective in preventing hot fluid from reaching the tubes. This high tube flow rate is also effective in transferring heat from the reactor vessel to the SG secondary side, thus helping to limit fluid temperature in the hot legs as well.

We believe a study would be beneficial of behavior with the SG secondary side depressurized after SG dry out. Now there would be no heat sink on the secondary side, and tube flow rates may be lower due to less of a driving force for natural convection flow in the SG. Further, we would expect to see further stratification in both the hot leg and the SG inlet plenum (the latter not being allowed in the PSNH supported analyses due to the modeling assumption of complete mixing). We pose the question of whether temperatures may be significantly above what was calculated by PSNH and its contractors under these conditions.

2. <u>Operator Induced Depressurization.</u> This calculation was based on the assumption that the operator would open an RCS PORV when the core exit thermocouples indicated 1200^{0}F. The calculations indicated accumulator discharge approximately 1400 sec after opening the PORV, with the RCS depressurized prior to vessel failure. The accumulators were emptied at about 10,600 sec, and vessel failure occurred 2000 sec later. Accumulator water was found to cause a small additional amount of hydrogen production. Phenomena associated with depressurization and hydrogen decreased the effectiveness of heat transfer between the core and other regions of the NSSS. Steam generator inlet plenum temperature reached a peak of roughly 850^{0}K during the depressurization, then cooled, and remained below 650^{0}K for the remainder of the calculation (20,000 sec total calculation time, with PORV opening at approximately 8000 sec). Maximum tube temperature was about 650^{0}K, and was reached at 20,000 sec, being identical to the SG inlet plenum temperature at that time. (Note RCS pressure is that of the containment following depressurization earlier in the calculation.)

We note that RCS pressure behavior (Ref. 18, Figure 4-4) is different for the base case and the PORV opening case prior to the time of opening of the PORV. We would like to discuss these differences for all parameters and we would like to understand the reasons they exist. (We note there is little difference in temperature over the range in question, and temperature is the important parameter for the SGTR issue.)

Volatile fission products represent about 20% of the decay heat, and the behavior of this energy source is calculated in the MAAP code. The calculations illustrated movement of the decay heat source. About 10% of the decay heat was associated with fission products which were in the upper plenum at the time of vessel failure. A small amount was in the hot legs, as was also the case for the pressurizer. The amount in the steam generator tubes was not significant. (Most of the CsI was in the upper plenum at the time of vessel failure, with about 10% of the CsI in the hot legs.)

3. Other Variations and Uncertainty. Several sensitivity calculations were performed to obtain a better understanding of behavior. These included:

a. Higher core melt temperature
b. RCP seal failure
c. SG secondary side blowdown
d. Core resistance variation
e. Reduced SG tube circulation
f. Core blockage changes.

These are discussed below.

a. Higher Core melt temperature. A case was run in which core melt temperature was assumed to be $3000^{\circ}K$ as contrasted to the base case $2500^{\circ}K$. This was intended to delay the onset of core geometry degradation, which in turn provides more time to heat other portions of the RCS. The $500^{\circ}K$ change in melt temperature was found to cause

only a few degrees change in SG tube temperatures, which was attributed to the extremely rapid temperature increase rate in the core as melt temperature is approached, and a concomitant small increase in the time available for heat transport to the steam generators.

The model is based upon assumed symmetric behavior, whereas some asymmetries have been found experimentally. If these contributed to a preferential flow of hot fluid near one of the hot legs, that leg might transport hot fluid toward a steam generator and provide higher temperatures than determined in the calculation. This could increase the computed impact of the sensitivity calculation.

A second aspect of the modeling that would act to reduce the calculated impact of the sensitivity run is the assumption of mixing within the steam generator inlet plenum. We believe an assessment of this effect is needed, as previously identified.

b. <u>RCP seal failure.</u> RCP seal failure, if it were to occur, was felt to be a leak in the range of 50 gpm (water) per seal. This was modeled, with the break occurring in all four RCPs at 45 minutes after initiation of the accident. This was found to have an insignificant impact on the results (Refs. 17 and 18).

PSNH also addressed preexisting leaks in SG tubes which are within technical specifications. These were stated to be small in comparison to the 50 gpm flow rate associated with seal leaks, and consequently were argued as being negligible (Ref. 17).

We believe the preexisting leak situation has a negligible impact on NSSS behavior as long as the leak remains small, but do not accept the argument advanced by PSNH as the reason. A comparison of the velocity associated with flow in a tube due to natural circulation with that associated with the leak, with establishing that the latter was negligible, would be more convincing. Similarly, a comparison of

flow rate induced by the RCP seal rupture to that expected for natural convection flow would be helpful. Further, one would have to establish that such a leak, passing steam, would not result in steam passing through the crossover leg seal at such a rate as to perturb the conclusions.

Provision of temperature information pertinent to fluid passing through the RCP seals would be helpful.

c. SG secondary side blowdown. Plys (Ref. 18) reports a calculation to investigate the effect of reduced cooling on the SG secondary side in which the steam generator PORVs are assumed to stick open, thus depleting the secondary side of a high pressure steam atmosphere. Drastic differences were discovered early in the accident due to cooling as the steam generators blew down. Sufficient cooling was provided that the pressurizer emptied due to primary fluid contraction. Reactor vessel failure occurred slightly earlier in this case as contrasted to the base case due to less heat removal from the primary system following removal of the secondary side heat sink. An initial peak in SG inlet plenum temperature of $860^{o}K$ is identical to that of the base case, but occurs about 500 sec earlier. Following the initial peak, the plenum temperature behavior is similar to the base case, although displaced in time, but is 50 to $100^{o}K$ higher over the remainder of the transient.

We suggest the calculation be conducted by assuming the PORV is stuck open after all water has been vaporized. This avoids the situation of overcooling associated with the early opening, and may be more compatible with some postulated operator actions associated with late attempts to deal with approaching core melt.

Again, we are concerned with the influence of assumed mixing in the steam generator inlet plenum and the impact upon calculated results.

d. Core resistance variation. Variation of the resistance of the core to flow was evaluated by lowering the axial and cross flow core friction factors in one calculation. This slightly increased heat transfer to the steam generators and correspondingly increased time to vessel failure. There was a slight tube temperature increase, but in general, the calculation showed little sensitivity of tube temperature to the change in core friction factors.

e. Reduced SG tube circulation. Selection of lower limit values of the number of steam generator tubes participating in flow from the inlet to the outlet plena was used for another sensitivity calculation. This provided lower values of steam generator natural circulation flow relative to the hot leg natural circulation flow rate, and reduced cooling of the steam generator inlet plenum due to flow from the outlet plenum. Slightly less heat was removed from the reactor vessel due to the lowered flow rates, and vessel failure occurred slightly earlier. These changes were insignificant. However, the steam generator inlet plenum was found to be about 150^{0}K higher than for the base case, reaching a temperature of 980^{0}K for a short time. Steam generator tube temperature was relatively unaffected.

Comparison of inlet plenum and tube temperature transient behavior (References 17 and 18's Figures 4-11 and 4-12) appears to indicate a significant thermal inertial associated with the tubes, which do not increase in temperature to a significant degree in contrast to the temperature of the source fluid in the steam generator inlet plenum. We believe this needs further discussion. For example, what is the location of the tube temperature and does this location correspond to the highest tube temperature?

Again, as previously stated, the influence of the assumption of complete mixing in the steam generator inlet plenum will impact the results. A portion of the concern is that reduced flow rates may

lead to greater stratification and less mixing in the SG plena, a phenomenon that is not modeled in the PSNH reported evaluations, and a phenomenon with the potential to increase tube temperatures over what was reported.

f. Core blockage. In this calculation, a delay of blockage in the core at the time of core melt to the time the node was completely filled with refrozen eutectic was assumed. This was done to continue core oxidation and core/upper plenum flow for a longer time. For this case, the maximum sustained SG inlet plenum temperature is roughly 1060^{o}K, with a short time (less than 50 seconds) temperature "spike" to about 1120^{o}K.

We again reiterate the concern with SG inlet plenum modeling and its impact upon the results.

g. Sensitivity Summary. An approximate comparison of the results of the sensitivity study is provided in Figure 1. The major early effect on increased tube temperature is due to changing the SG tube flow characteristics. Later, and with the greatest impact, is the effect of delaying formation of blockage in the core, which allows continued circulation of hot fluid through the core where the temperature is increased, as opposed to a drastic reduction in heat transport between the core and other RCS components when a core geometry change occurs.

4. Steam Generator Tube Strength. Plys, in Reference 18, Appendix B, addresses SG tube integrity. The presentation is based upon the SG secondary side pressure being at the SG safety or relief valve setpoints which, as previously discussed, may not be the case. We note that Plys identifies nominal hoop stresses of 9300 to 10000 psi for the assumed conditions. Hence, the case of the SG secondary being depressurized will

result in a nominal hoop stress of roughly 19,000 psi. This stress, substituted into Reference 18's Figure B-6, results in a Larson Miller parameter of about 37. The Larson Miller parameter is defined as:

$$LM = T(20+\log t_r) \times 10^{-3}$$

where:

T = temperature, ^{o}R

t_r = time to rupture, hrs.

Substituting a temperature of 1090^{o}K (the value used by Plys to conclude the rupture time would be greater than 2.5 hrs) yields a time to rupture of about 5 minutes, a significant change from the Plys value.

Plys could have selected 1090^{o}K as conservative, with no need to consider an alternate since the no tube rupture position was supported by the result. If we recognize this possibility, and select a less conservative 1000^{o}K, we find a rupture time of about 3.5 hours. These temperatures can be contrasted to the SG inlet plenum temperatures provided in Figure 1, with recognition that these are not tube temperatures, but also with the recognition that some of the parameters contributing to the temperatures remain to be evaluated.

Clearly, we are in a temperature region where relatively small changes have a significant impact upon creep rupture time. Equally clearly, tube stress could be roughly a factor of two higher than the value used to justify that tubes would not rupture. We conclude the picture is not as clear as presented in Reference 18, which presented a conclusion that tubes would not be ruptured.

3.2.8. Other Considerations

In Reference 17, PSNH stated that if one postulated creep rupture failure of steam generator tubes, the pressure inside the previously dried out and isolated steam generator secondary side would increase until the steam

generator PORV's setpoint was reached, at which time the valves would lift and modulate until reactor vessel melt through and RCS depressurization into the containment. During the periods of SG PORV opening, there would be a high leak rate bypass condition directly from the RCS to outside the containment. They further stated that after vessel melt through, the leak rate out this path would be low and would correspond to any low pressure leakage through the reclosed PORV. They note this leak path could be enhanced if the SG safety valves also lift and fail to reseat properly; however, they believe it unlikely that the safety valve setpoint would be reached.

As previously discussed, we do not believe an individual tube would rupture, but instead believe there would be a massive failure in one steam generator. (Once the failure initiated, we would expect the RCS to depressurize rapidly, which would reduce stress on tubes in other steam generators.) It is difficult to postulate a PORV modulating this condition. It is further difficult to postulate the PORV or the safety valves would not be damaged when exposed to these conditions, and therefore their reclosing may be questionable. One may also question SG secondary side structural integrity when exposed to the high temperature environment. Finally, if the conditions which led to the accident sequence involve a loss of all AC power, which is one of the likely situations given a severe accident scenario, we pose the question of how long the PORVs can be expected to modulate pressure assuming they are not damaged by the fluid being modulated.

Plys (Ref. 18) has identified that the MAAP code does not model certain aspects of SG tube temperature, and a method of obtaining temperature was discussed. Aside from the impact of secondary side steam as a cooling medium, we are concerned about local heating due to small leaks. Such a leak could cause a small amount of hot fluid to pass through a localized area into the SG secondary side, with different heat transfer characteristics and tube temperatures than one would encounter with the treatment of overall inside to outside heat flow utilized by Plys in their estimation. Whether this is important to localized tube temperature over a

sufficient area to be of concern should be addressed. (Note the effect could also be concentrated in an adjoining tube. This can be visualized by picturing a tube with a small hole which directs hot RCS fluid onto the secondary side surface of an adjoining tube, while the inside surface of that same tube is exposed to hot RCS fluid.)

3.3 Accident Likelihood

PSNH has estimated the mean annual frequency of accidents in which the core melts with the RCS at high pressure and the SGs dry as bounded by a value of 4.5×10^{-5} per reactor year (Ref. 17). This is composed of the following plant damage states:

Plant Damage State (PDS)	Mean Annual Frequency
3D	1.5×10^{-5}
3FP	8.9×10^{-6}
4A	1.4×10^{-5}
4C	1.7×10^{-7}
4D	2.8×10^{-6}
4E	2.2×10^{-11}
4FP	1.2×10^{-7}
8A	3.9×10^{-6}
Total	4.5×10^{-5}

The accident sequences which comprise the PDSs include transient and loss of off site power sequences with failure of all emergency feedwater, failure of feed and bleed with loss of all emergency feedwater, and transients without scram. PDS 8A consists of eight sequences which involve station blackout and emergency feed water failure with recovery of containment heat removal.

PSNH also addresses the potential impact of tube rupture on this information. They have assigned a high chance of no containment failure to PDSs A. PDSs C and D are considered as leading to a high likelihood of long term containment overpressure failure. PDSs FP are a high chance of small bypass, and PDS E is

a high chance of large bypass. Hence, PDSs A, C, and D would be impacted by SGTR, and FP may represent some impact. Addition of the appropriate values indicates that the likelihood of being in a condition where SGTR could affect the results is about 4×10^{-5} (as contrasted to the assumption of no SGTR).

PSNH considers these values to be bounding because some of the values include states with water on the steam generator secondary side, for which SGTR is not a concern, certain operator recovery actions have been neglected, and RCS depressurizations prior to core melt have not been considered. As previously discussed, operator depressurization is one of the potential steps which one could consider to mitigate SGTR. PSNH estimates the frequence of operator failure to depressurize as less than 10^{-2} to 10^{-3} per demand, provided procedures are modified and adequate operator training is provided. Using these values results in a frequency of obtaining conditions under which SGTR would be of concern of about 10^{-7} to 10^{-8} per reactor year.

Although these values appear reasonable, we note that the conditions which led to the factor of 10^{-2} to 10^{-3} reduction do not presently exist. We further would need substantiation for these values prior to acceptance.

Discussion is also provided concerning the likelihood of SGTR if exposed to high pressure core melt conditions (Ref. 17). PSNH points out that their calculations show SG tube temperatures that are roughly 200 to 300^{0}F below what would be required for creep rupture, and this is identified as principally due to cooling by steam on the SG secondary side. Several things are necessary for acceptance of the tube temperature conclusions, including, as discussed elsewhere, substantiation of the calculational technique and investigation of the likelihood of the SG secondary side having a significant steam inventory (which also means having a significant pressure).

Finally, PSNH estimates a 99% chance that failure of SG tubes will not occur before reactor vessel melt through or piping nozzle failure. This value, combined with the prior PSNH estimates of frequencies, appears sufficient to

establish that SGTR is not of concern as a significant contributor to risk. Therefore, one can reasonably anticipate that substantiation of the various items which led to the conclusion, as discussed in this communication, should provide substantiation of the above preliminary conclusion.

3.4 Additional Observations

A number of observations and comments have been made in the previous discussion. We offer the following additional comments:

1. Much of the modeling utilized in the calculations has not been documented. We understand this is underway. Such documentation will be helpful in the continuation of the review.

2. The outside of the hot legs is assumed to be adiabatic. This probably introduces a small conservatism into the results with respect to hot leg temperature. The impact on other parameters is probably negligible. With respect to the hot legs, the parameter of interest may involve a relatively thin wall connecting pipe that is exposed to high fluid temperature, and whose temperature will follow fluid temperature more closely than is the case with the relatively massive hot leg: or the vessel nozzle region of the hot leg, which will be more closely allied with fluid circulating rapidly within the upper plenum. Thermal response of these regions may be critical in determination of the failure point of the RCS pressure boundary.

3. Although the limited experimental evidence reveals some symmetry in flow behavior within the reactor vessel, there are also unsymmetrical flows and temperatures. We understand the MAAP calculations are based upon modeling the upper plenum fluid as a single volume. This appears to be a nonconservative approach.

4. STEAM GENERATOR TUBE RUPTURE CONCLUSIONS

The above discussed considerations lead us to the conclusion that this topic is in a developing state, with knowledge being rapidly accumulated. Insufficient information is presently available for one to conclude that SGTR cannot occur as a result of severe accident conditions.

Our judgement, at this juncture, is that a carefully conducted and thorough evaluation on the part of PSNH, that utilizes information which either exists or will be available within the near future, can establish that the likelihood is small that a SGTP will result due to overheating during severe accidents. Further, our judgement is that the risk associated with SGTR can be shown to be negligible for these conditions. Our judgement needs to be substantiated. We have encountered too many unanswered questions, unsubstantiated assumptions, and potential conditions which could lead to calculation of increased temperature to accept a conclusion that SGTR will not occur under circumstances such that the associated risk can be neglected. We further judge that coverage of all areas subject to question will be a substantial task. We note, as a qualifier to these conclusions, that our review is not complete, and, in addition, work is ongoing to provide further information.

Existing knowledge would support a conclusion that SGTR is not a problem if the RCS is depressurized. Consequently, reasonable assurance that progressions toward core melt would not occur at high RCS pressure, coupled with suitable technical backup for a conclusion that low pressure is not of concern, would eliminate our concern regarding SGTR under severe accident conditions. We have not conducted an evaluation of the trade-offs associated with such an approach, nor have we been provided with information that would either support or negate RCS depressurization under severe accident conditions. We have not provided a recommendation regarding whether RCS depressurization is attractive when all pertinent factors are considered due to lack of a balanced picture.

Determination of the correctness of a judgement that SGTR is not a concern under severe accident conditions with the RCS at high pressure can be based upon a combination of analytic and experimental investigations. The ongoing test at Westinghouse in which reasonably close similitude is claimed between

the test facility and appropriate parts of a Westinghouse four loop NSSS should provide key data which can be applied to assist in the confirmation of analysis techniques. Selected test data from other facilities and further examination of the analysis techniques, coupled with necessary changes when they are uncovered, should provide sufficient confirmation that reasonable reliance can be placed upon accident analyses pertinent to this issue. Application of a reliable analysis technique to issue investigation should then provide the necessary background to resolve this issue. Such a program will represent a formidable undertaking.

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FEB 03 1987

MEMORANDUM FOR: Vincent S. Noonan, Project Director
Project Directorate #5
Division of PWR Licensing-A

FROM: Charles E. Rossi, Assistant Director
Division of PWR Licensing-A

SUBJECT: SEABROOK EMERGENCY PLANNING STUDY -
TREATMENT OF PREEXISTING LEAKS IN CONTAINMENT

As a part of the staff evaluation of the applicant's submittal on the Seabrook Station Emergency Planning Zone, the treatment of preexisting leaks regarding the containment isolation dependability was reviewed by the staff in the Engineering Branch. Attached is a draft evaluation of the treatment of preexisting leaks in containment.

This evaluation concludes that (1) the Seabrook purge and vent valves in a fully closed configuration should be capable of withstanding the severe accident induced pressure, and (2) the applicant has presented a reasonable approach for considering preexisting leaks in the containment.

Original signed by
Charles E. Rossi, Assistant Director
Division of PWR Licensing-A

Attachment: As stated

cc: T. Novak
R. Ballard
S. Long
V. Nerses
S. Newberry
G. Bagchi

Contact: G. Bagchi
X27070

ATTACHMENT 1

SEABROOK STATION EMERGENCY PLANNING ZONE STUDY EVALUATION OF TREATMENT OF PREEXISTING LEAKS IN CONTAINMENT

Background: Demonstration of operability of the containment purge and vent valves against internal pressure from a design basis accident is required to assure dependability of containment isolation. The safety evaluation of the operability qualification is documented in NUREG-0896 Supplement Number 5, Appendix Q. The staff obtained the basic information on the Seabrook containment purge and vent valves as a part of the licensing review under the TMI Action Item II.E.4.2. In order to assess the behavior of these valves in the severe accident environment, the staff has used this basic information on the valves, and assumed that the valves would be fully closed during the severe accident phase because of their demonstrated ability to close under the design basis accident condition.

Also, based on numerous reports from various licensees on unavailability of containment function and reports of failures of type C leak rate tests of containment isolation valves, it is important in any risk analysis to take into account the effect of preexisting leaks that may have gone undetected during the plant operation prior to a postulated severe accident. Therefore, a study of unavailability of containments was undertaken under NRC sponsorship by the Pacific Northwest Laboratory (PNL). PNL reported the findings of its study in NUREG/CR 4220, "Reliability Analysis of Containment Isolation Systems." This study estimates the probability of larger leaks (28 square inches) to be in the range of 0.001 to 0.01 with a point estimate of 0.005. During its review of the Seabrook Emergency Planning Zone (EPZ) study, the staff requested additional information from the applicant to address the effect of preexisting leaks in its assessment of the probability of various release categories.

The purpose of this evaluation is to (1) document the staff assessment of the capability of the Seabrook purge and vent valves to resist the severe accident environment and (2) to determine the reasonableness of the applicant's approach for the consideration of preexisting containment leaks.

Evaluation: As reported in NUREG-0896 Supplement Number 5, the Seabrook purge and vent valves are 8 inch butterfly type Posi-Seal (Model 2898B), Class 150 with Matryx airactuator (Model 26062-SR60). There is a pair of valves in each flow path with independent flow interruption capabilities and on loss of air the valves close due to spring loading. These valve assemblies are analyzed for seismic loading of 3g per axis with loads along all axes acting simultaneously and superimposed aerodynamic load simulating the pressure load from a design basis accident. The combined stresses are kept under the ASME Code allowable values. The valve seat material is resistant to containment spray chemicals and radiation. The 1-year accident dose rate is calculated to be approximately 1.2 x 10^6 rads compared to the material resistance level of 10^8 rads. These valves also have screens in elbows upstream of the valves to stop debris from entering the valve seating area.

The Posi-Seal 8" Class 150 wafer-type butterfly valves have an ANSI rating of 230 psig at 300°F i.e. a capability to hold against a pressure of 230 psig at a temperature of 300°F. The highest stresses due to the 3g seismic and combined design basis accident pressure of 60 psig are as follows:

Valve stem	23,331 psig	(52,500 psig allowable)
Disc pin	21,699 psig	(52,500 psig allowable)

Based on the above discussion these valves are capable of resisting the containment capability pressure of 157 psig and the pressure of 180 psig at 1% hoop strain including the temperature associated with the wet containment condition along with the expected radiation exposure.

The applicant in its letter dated October 31, 1986, responded to the staff request for additional information number 22. In the original study (PLG-0300) the applicant quantified preexisting containment leaks at the rate of 0.1% per day for the release category S5, and with all other release categories estimated the effect of containment failures and bypasses including failure to isolate the containment. It is noted in the applicant's response that the containment purge and vent valves at Seabrook are leak tested every six months or less and their position is checked monthly. Also, manual isolation valves outside containment are position checked every month. Thus the large pre-existing leakage with a probability estimate of 0.01 to 0.001 in NUREG/CR-4220 may not be appropriate for Seabrook.

In spite of the specific differences at the Seabrook Station, the applicant considered the effects of both small and large preexisting leaks in its EPZ study. For the small preexisting leakage the applicant estimated that a rate of ten times the allowable leakage would yield zero early fatalities and a small contribution to early injuries. For a large leakage, assumed to be a six inch valve (on 28 square inch hole), with a conditional probability of 5E-3 from NUREG/CR-4220, the applicant estimated the health impacts using an S6W release category. Their estimate, which they believe to be conservative, is attached as Figure-1.

Conclusion: Based on its review of the information available the staff concludes that the purge and vent valves in a fully closed configuration should provide reliable isolation of the Seabrook containment under severe accident conditions up to the pressures corresponding to 1% hoop strain in the containment.

The staff also concludes that the applicant has presented a reasonable approach for the consideration of preexisting leaks, both small and large.

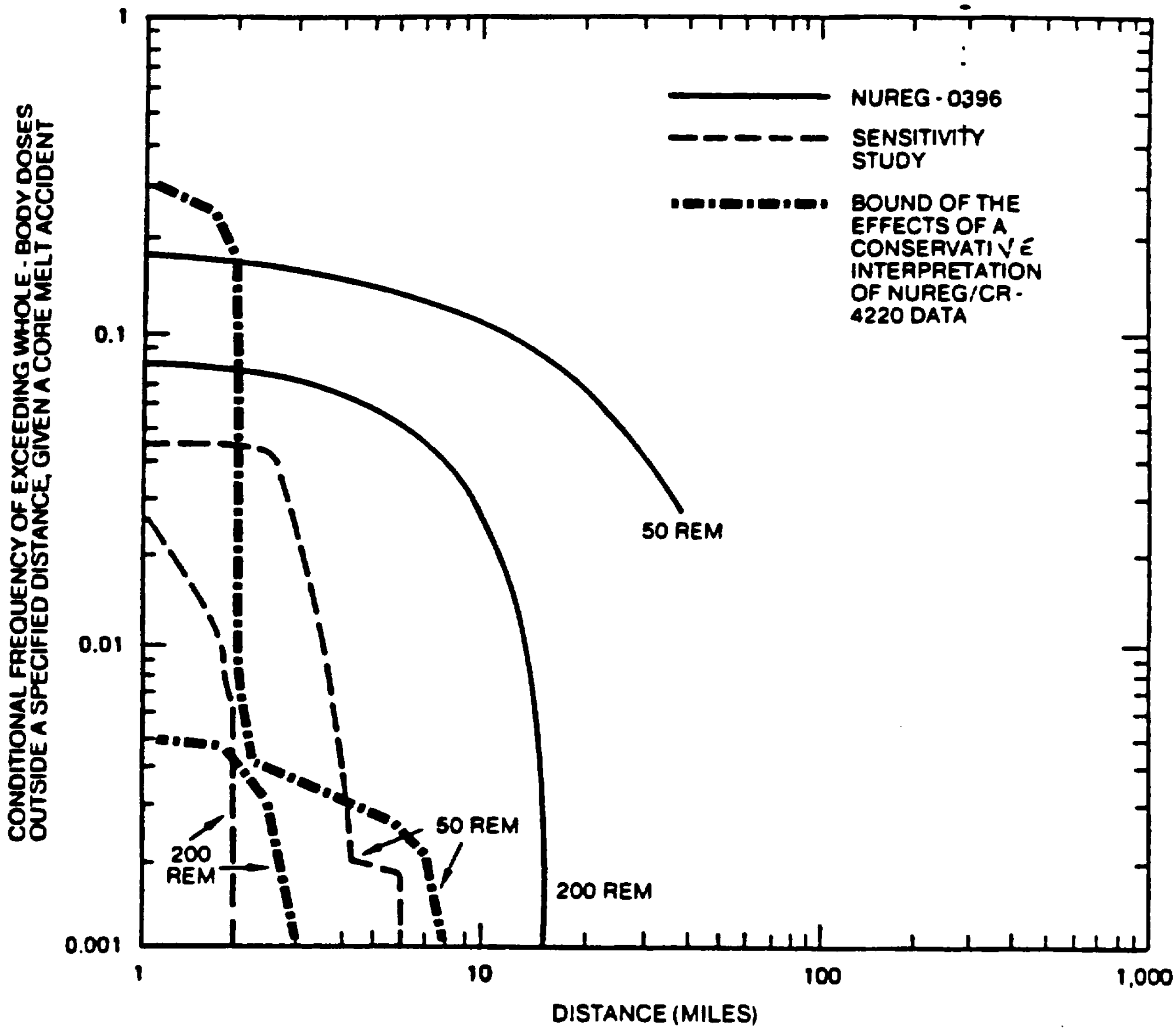

Figure 1 BOUNDING THE CONSERVATIVE INTERPRETATION OF NUREG/CR - 4220 DATA PRE-EXISTING CONTAINMENT LEAKAGE

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The characterization of uncertainty in Probabilistic Risk Assessments of complex systems

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This paper discusses the issue of the characterization of uncertainty in a Probabilistic Risk Assessment (PRA) of a complex system, such as a nuclear power plant. The significance to the interpretation of the results of a PRA of maintaining the distinction between the aleatory and epistemic components of uncertainty is illustrated using a simple example. The point of view presented here is that the degree to which it is necessary to invoke both aspects of uncertainty to characterize an event in a PRA model is as much a function of the way the analyst chooses to model the event of interest as it is of the nature of the event itself. © 1996 Elsevier Science Limited.

1 INTRODUCTION

As discussed by the guest editors in their request for contributions to this special edition of *Reliability Engineering and System Safety*, it is becoming increasingly important to decision makers that, when presented with the results of a Probabilistic Risk Assessment (PRA) of the mission being performed by a complex system, the uncertainty in the results of the PRA is correctly characterized. PRA studies are being performed for space missions, chemical processing facilities, and waste storage facilities, and have been performed for the majority of the nuclear power plants (NPPs) in the USA, and a large number overseas. For many of these studies, the organization commissioning these studies requires a discussion of the uncertainty in the results. The characterization of uncertainty in the numerical results has been a feature of NPP PRAs since the publication of WASH 1400,[1] one of the earliest large scale PRAs to be completed. The representation of uncertainty was elevated to greater prominence in the Zion[2] and Indian Point Probabilistic Safety Studies.[3] The philosophy behind the approach to uncertainty adopted in these latter two studies is described in the paper by Kaplan & Garrick.[4] The PRA Procedures Guide,[5] NUREG/CR-2300, written to provide guidance for analysts performing PRAs of nuclear power plants, dedicated a chapter to the discussion of uncertainty and sensitivity analysis. Uncertainty was categorized into three types; parameter uncertainty, modelling uncertainty, and completeness uncertainty. The representation of parameter uncertainty, and its propagation to characterize the uncertainty on the numerical output of a PRA was discussed at length, but relatively little attention was given to the other two types. While, as discussed later, it can be argued that the distinction made in the PRA Procedures Guide between parameter uncertainty and modelling uncertainty is fundamentally artificial, it is certainly useful from a practical point of view. Parameter uncertainty can be thought of as addressing uncertainty in the quantification of a model with a specified functional form, whereas model uncertainty can be thought of as addressing the uncertainty in the appropriateness of the structure or mathematical form of the model. Completeness uncertainty is a special category of model uncertainty that is associated with the degree to which the model addresses all the phenomena associated with the system being modeled. Whether it is possible, or even makes sense, to try to capture completeness uncertainty formally, however, is open to question, and will not be addressed here.

Another categorization of uncertainty is that which the guest editors have requested the contributors to this special edition of *Reliability Engineering and System Safety* to address, namely the categorization of uncertainty as either being of an aleatory or an epistemic nature. The terms aleatory and epistemic have only recently been introduced into the literature

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of NPP PRAs (see, for example, Ref. 6). The aleatory aspect of uncertainty is that addressed when we characterize the events or phenomena being modeled as occurring in a 'random', or 'stochastic' manner, and adopt probabilistic models to describe their occurrences. It is this aspect of uncertainty that gives the Probabilistic Risk Assessment the probabilistic part of its name. The epistemic uncertainty is that associated with the analyst's confidence in the predictions of the PRA model itself, and is a reflection of his assessment of how well his model represents the system he is modelling. In this sense, the uncertainty that was addressed in Chapter 12 of the PRA Procedures Guide[5] was epistemic.

The point of view adopted in this paper is that it is essential to maintain the distinction between these two types of uncertainty[7,8] as they perform different functions in the model of the system created by the analyst. The aleatory uncertainty is a fundamental and integral part of the structure and form of the PRA model, whereas the epistemic uncertainty is related to a characterization of how well we can represent the system by the model. In practice, however, many analysts have found that, for certain issues, especially those related to the modelling of the occurrence or the impact of particular physical phenomena, particularly in regimes that are outside our direct experience, it is difficult for them to distinguish between the two types. It is this writer's belief[8] that the confusion has been exacerbated because the same mathematical tool, probability theory, is used to parameterize and quantify both types of uncertainty. It has not been uncommon for analysts to avoid addressing the issue by claiming that the distinction is irrelevant. However, as discussed in this paper, it is important to distinguish between the two, not only because it can impact the answer being given to a decision maker, and hence have an impact on the decision outcomes, but because it is essential to truly understand the nature of the model of the world that is being incorporated in the PRA.

When it has been accepted that it is important to maintain the distinction between the two types of uncertainty, it is not uncommon to hear analysts ask whether the uncertainty associated with a particular model element is epistemic or whether it is aleatory, as if the type of uncertainty were a property of the issue being modeled. The situation in many, if not most, of the interesting cases, is that, in modelling the issues, an analyst could make the case for both types of uncertainty. Therefore, as discussed later, the question that should be asked by the analyst is, 'how am I modelling this issue?' Understanding the modelling process is the key to an appropriate representation of uncertainty, and hence, ultimately, to making an appropriate use of the results.

The next section discusses what we mean by a model in the context of this paper, and why, for models in general, and PRAs in particular, it is necessary to address uncertainty. Following that, the treatment of uncertainty in PRA models is briefly reviewed. The subsequent sections of the paper discuss, in turn, the thought process an analyst should pursue to clarify the meaning of the models of the constituent elements of the PRA, and the importance of explicitly keeping track of model uncertainty in order to correctly interpret the result of a PRA addressed.

2 MODELS AND UNCERTAINTY

A model can be described as an analyst's attempt to represent a system (using the term system in a very general way) in a form that can be used as an explanatory and an exploratory tool. It is, in almost all cases, impossible to capture all the subtleties of the system behaviour and, therefore, any model is, at best, an approximation. A model in the physical sciences or engineering disciplines is usually a mathematical model, which is to say that it has a mathematical form which can produce numerical results that represent some observable aspects of system behaviour. Such a mathematical model will generally have one or more parameters. Since any model is an approximate representation, it follows that there must be some (epistemic) uncertainty associated with the formulation, and predictions, of the model. For some models, however, this uncertainty is so small that it can essentially be ignored. For example, the mathematical formulation of many of the models created by physicists to explain natural phenomena are sufficiently well supported or verified that the models are very precise in their predictions, within a specified region of applicability. In addition, many of the parameters are so well known that they can be thought of as universal constants. An example of one such model is Newtonian mechanics and Newton's law of gravity, which is capable of making very accurate predictions of such things as planetary motion, and can be used to define the trajectories of planets or space vehicles with great accuracy. Not only is the model rather simple but the parameter of the model, the gravitational constant, is known very accurately. Of course, it is well known that, under specific boundary conditions, and for particular problems, Newtonian mechanics breaks down and must be replaced with the General Theory of Relativity. Newtonian mechanics is an example of a deterministic model. A model need not, however, necessarily be deterministic to be precise. Quantum Electrodynamics (QED) is a model which is capable of making very accurate predictions. However, because of the quantum mechanical nature of matter in the small scale, it does so only in a probabilistic sense, making

of NPP PRAs (see, for example, Ref. 6). The aleatory aspect of uncertainty is that addressed when we characterize the events or phenomena being modeled as occurring in a 'random', or 'stochastic' manner, and adopt probabilistic models to describe their occurrences. It is this aspect of uncertainty that gives the Probabilistic Risk Assessment the probabilistic part of its name. The epistemic uncertainty is that associated with the analyst's confidence in the predictions of the PRA model itself, and is a reflection of his assessment of how well his model represents the system he is modelling. In this sense, the uncertainty that was addressed in Chapter 12 of the PRA Procedures Guide[5] was epistemic.

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predictions about the average behaviour of a population of events rather than about the outcome of a particular event. It is therefore a probabilistic or aleatory model of the world.

The models that go to make up a PRA, by contrast with Newtonian Mechanics and QED, are considerably less well established. Furthermore, since a PRA is used to model very rare events, there can be no experimental verification of its validity. In addition, because of the rare nature of the events being modeled, statistical uncertainties in the estimates of the parameters of the model can be significant. Furthermore, and perhaps of most interest here, there are uncertainties about the impact of physical phenomena taking place during accident scenarios that create differences of opinion about how to model these impacts. Thus, as discussed in more detail in the next section, there are considerable uncertainties associated with creating a PRA model, even at the level of the individual elements of the model.

The uncertainty associated with modelling these elements could be in the choice of mathematical form of the model, it could be in the values of its parameters, or it could be in both. To the extent that changes in the parameter values are little more than subtle changes in the form of the model, it could be argued that there is really no precise distinction between model uncertainty and parameter uncertainty. However, the characterization of uncertainty in parameter values for a given model, at least for several models with specific functional forms, can be performed in a compact mathematical way using the subjectivist interpretation of probability and the tools of Bayesian statistics.[9] In this case, the range of possible values the parameters can take is continuous. Methods of propagating the uncertainty on the parameter values to characterize the uncertainty on the predictions of the model are well established (see, for example, Ref. 10) and have been applied in many PRA studies. A probabilistic characterization of uncertainty in the form of a model can be performed by generating a probability distribution over a discrete number of plausible models with the probabilities representing the analyst's degree of belief as to which model best represents the system or phenomenon being modeled. While this may appear to be a simple proposition, an explicit treatment of the uncertainty in the form of the model is not common in PRAs. In the next section, the nature of the PRA model is discussed in relation to the treatment of uncertainty.

3 UNCERTAINTY IN PRA MODELS

There are three elements to performing an uncertainty analysis in a model like a PRA: characterizing the uncertainty on the individual elements of the model; propagating these uncertainties to obtain a characterization of the uncertainty on the output of the model; and interpreting the results in light of the uncertainty.

3.1 Uncertainty in the characterization of the basic events

A PRA is based upon logic structures such as event trees and fault trees, that identify the different combinations of more elementary events, called basic events, that could lead to undesired system states. The types of basic events found in PRAs include events such as: the failure of a pump to start, the failure of a pump to run for 24 hours, the occurrence of an initiating event such as a reactor trip, failure of an operator to take the appropriate actions to prevent system damage. The majority of these basic events are regarded as resulting from random processes and are described by probabilistic models.

The probabilities of events generated by these probability models essentially represent the relative fractions with which various outcomes would be expected given a population of identical replications of the system of concern were hypothetically to be observed a large number of times.[11] What these fractions represent is not necessarily an 'inherent' randomness in the system behaviour, but the fact that, at the level at which the basic events are defined, there are hidden variabilities that are accommodated in the model that way. There are variabilities in underlying conditions that would have an impact on the behaviour of individuals in the population that are not being explicitly accounted for. Instead, their average impact is implicit in the probability models used for the basic events. Thus, the relative fractions are parameters of a model of the world[6] in which groups of components are regarded as being members of the same population. It is important to remember that there are individual characteristics of the members of the population for which the probability model is constructed that are not explicit in the formulation of the model. A different model of the world can lead to different sets of variables being suppressed, different definitions of basic events, and different probability models.

For many of the basic events of the PRA model, the associated probability models are simple, with only one or two parameters. An example is the simple constant failure rate reliability model, which assumes that the failures of a component while it is in standby occur at a constant rate. The parameter(s) of such models can be estimated using appropriate data, which, in the example above, comprises the number of failures observed in a population of like components in a given time. In most recent PRAs, the parameter estimation has been accomplished by adopting a Bayesian or subjectivist framework[9] which uses

probability as an index of the analyst's assessment of the appropriateness of the possible values of the parameters, thus representing an epistemic uncertainty. This epistemic uncertainty is represented by a continuous probability distribution on the value of the (aleatory) parameter. Thus, PRA models typically address two types of uncertainty, the aleatory uncertainty that results from the adoption of the concept of randomness as a means of capturing variability in underlying conditions that is not explicitly modeled, and the epistemic uncertainty which characterizes the analyst's knowledge about how to parameterize this variability. At this level, there has been relatively little difficulty in communicating the different aspects of uncertainty as long as analysts have accepted the use of subjective probability as a means of characterizing uncertainty. Some statisticians who adhere to the classical school of thought have found this difficult to accept (see, for example, Ref. 12).

There are other basic events of the PRA model, for which it is accepted that the appropriate representation is as random events, but for which there is no single generally accepted probability model. A particularly well known example is the set of events that represent the occurrence of earthquakes of different magnitudes, in particular when the magnitudes are beyond the range of current experience. These frequencies of these events are obtained from seismic hazard curves. There are several models that can be used to create the seismic hazard curves, and each one is complex, and based on many assumptions. It has become customary in this case to produce a set of hazard curves, corresponding to different sets of assumptions, and use the results of these models to represent the range of values of the frequency of the occurrence of the event. This is an example of an explicit representation of the uncertainty in seismic hazards as a family of curves, which translates into a discrete distribution on the parameter characterizing an event in the model, i.e., the earthquake occurrence frequency. In many cases, probabilities have been assigned to the members of the family of curves to represent the analysts belief in each of the curves as the most appropriate representation of the hazard.[2,3,13] The results of the analysis of seismic hazard can then be regarded as a discrete probability distribution on the value of the earthquake frequency, an aleatory parameter.

However, there are events for which there may be no well established models with which to estimate the probabilities. These events are often associated with the representation of the occurrence or not of a specific phenomenon, and particularly arise in what is called the level 2, or containment response portion, of a PRA. An example of such an event is containment failure due directly to a steam explosion. In a paper on the analysis of containment failure due to a steam explosion following a postulated core meltdown in a light-water reactor,[14] the authors recognized that estimating the probability of this event was a very difficult task. Furthermore, they recognized that the interpretation of an assessment of this probability that had been made in the Reactor Safety Study[1] was difficult to determine, asking: 'Does this probability reflect a stochastic process in which 1 in 100 core-meltdown accidents would involve containment failure by steam explosion, or is this a measure of the uncertainty of the phenomenon?' However, in many instances, the analysts concerned have indeed adopted a position with respect to the interpretation of such probabilities, and have declared them to represents either an aleatory or an epistemic uncertainty.

3.2 Propagation of uncertainty

Methods for the propagation of the uncertainty on the basic events through the quantification process, to generate a characterization of uncertainty on the output of the PRA, are relatively well established.[10] Because epistemic uncertainties on parameters are generally characterized as probability distributions, whether the distributions are continuous or discrete, the most common technique is Monte Carlo analysis or variants thereof, such as the Latin Hypercube Sampling. If all the parameters associated with the basic events represent an aleatory property, the process is straightforward. However, it is natural for an analyst constructing an event tree model to include events directly in the event tree logic model structure, even when their probabilities are deemed to represent epistemic uncertainty, because the event tree is in essence only a delineation of possible sequences. However, as discussed in more detail later, the quantification process, and particularly the uncertainty analysis, must take account of this difference, for reasons elaborated on in Section 5 of this paper. This was recognized in some PRAs in the early 1980s,[15,16] and, more recently, in the PRAs performed for the USNRC in support of NUREG 1150,[17] and that performed for the La Salle PRA.[18] The approach to the analysis for the latter two studies is described in Ref. 19. The probabilities associated with the branches on the event trees that are considered to be aleatory in character are multiplied together to generate sequence probabilities. The probabilities associated with the branch points that represent epistemic uncertainty do not contribute to the sequence frequencies. Instead, they are used when performing the uncertainty analysis, to determine the relative fractions of the Monte Carlo samples (Monte Carlo methods are the most commonly used methods for propagating uncertainty in PRAs) in which the paths through the branch point appear. The evaluation and

probability as an index of the analyst's assessment of the appropriateness of the possible values of the parameters, thus representing an epistemic uncertainty. This epistemic uncertainty is represented by a continuous probability distribution on the value of the (aleatory) parameter. Thus, PRA models typically address two types of uncertainty, the aleatory uncertainty that results from the adoption of the concept of randomness as a means of capturing variability in underlying conditions that is not explicitly modeled, and the epistemic uncertainty which characterizes the analyst's knowledge about how to parameterize this variability. At this level, there has been relatively little difficulty in communicating the different aspects of uncertainty as long as analysts have accepted the use of subjective probability as a means of characterizing uncertainty. Some statisticians who adhere to the classical school of thought have found this difficult to accept (see, for example, Ref. 12).

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interpretation of a point estimate requires some care. As pointed out in Ref. 19, while in principle, maintaining the separation is rather straightforward, it is deciding on how to interpret the probability associated with a particular issue that is often the problem. Reference 20 presents an example of how that decision was made for the study described in Ref. 16. This issue is discussed in more detail in the next section.

3.3 Interpretation of the results of a PRA

Interpreting the significance of the results of a PRA in the light of the uncertainties is important if the PRA results are to be applied to making meaningful decisions about changes in design or operating practices, or if they are to be used for economic decisions. Probability distributions on the numerical results, such as the core damage frequency, can be used to calibrate the confidence level at which a safety goal is being met for example. However, while it may be important to characterize the overall uncertainty, it is equally important to understand which factors drive the uncertainty. When modelling uncertainty is included in the PRA, it is essential to be able to distinguish between the results of the alternate models.[20] Correctly representing and propagating the epistemic uncertainties facilitates this as illustrated in Section 5. While the example in Section 5 is a simple one, the current generation of PRA models can be complex. For the comprehensive models generated for NUREG 1150, special computer techniques have been developed (see, for example, Ref. 21) to track the influence of uncertainties.

4 DETERMINING THE NATURE OF UNCERTAINTY

In order to discuss the issue of how to determine the nature of the probability associated with an element of a PRA model, we take as an example that discussed above of the containment event tree branch point that represents the occurrence or not of a steam explosion large enough to fail a containment. The purpose of this example is not to provide a discussion of the physics and engineering aspects of the issue. Instead, this example is used to illustrate how, by taking time to understanding the analysis process, an appropriate characterization of the uncertainty type can be determined. As a starting point, it will, for the purposes of this discussion, be assumed that there are two important elements to analysing this situation, namely an assessment of the potential energy yield, and an assessment of the strength of the containment. Suppose, for the sake of argument, that the energy yield is considered to be a function of the detailed history of the scenario that led up to the core damage. In a PRA defined scenario, this detailed history is not explicitly represented. For example, while the failures that contribute to the scenario are defined explicitly, their timing is generally not. As another example, if the scenario being modeled is initiated by a LOCA, for the purposes of defining success criteria, a LOCA of a certain size may have been chosen to represent a range of break sizes. Because the underlying models are based on a description of the world as exhibiting random or variable behaviour, the boundary conditions of the level 2 (containment performance) analysis provided by the end points of the level 1 (core damage) analysis are not determined uniquely, but encompass a range of conditions. The core damage or plant damage state sequences in reality represent classes of possible real world scenarios, of which the individual members may vary in many aspects. Thus, the answer to the question about how to represent the events on the containment event trees should be evaluated taking this variability into account. Suppose that the relationship between the hidden variables and the yield were known, that it is possible to investigate this variability, and further that the strength of the containment is known accurately. Then, under these conditions, the 'likelihood' of the branch point could be estimated as the relative fraction of the realizations of that sequence which lead to a yield that exceeds the strength of the containment. This estimate of the 'likelihood' would then be a parameter of the model of the world that represents the aleatory aspect of the model in that it addresses the variability implied by that model.

However, the elementary reliability models used to generate the plant damage states may not support addressing the underlying variability in the degree of detail necessary (because detailed knowledge about the causes of variability is suppressed in probabilistic models), and the analyst may find assessing this fraction extremely difficult. An alternative approach might, therefore, be to try to determine whether the conditions that would lead to the undesired outcome could ever arise. In this case, a bounding analysis of the energy yield from the steam explosion and an assessment of the strength of the containment would be needed. If the bounding analysis demonstrated that the containment strength would not be exceeded then the event can be said not to occur, i.e., its 'likelihood' would be zero. If, however, there are epistemic uncertainties in the inputs to the calculations, for example, in the evaluation of the containment strength, then the answer may not be so clear cut. For example, as was the case in the analysis discussed in Ref. 20, assume that an analysis has resulted in the characterization of the uncertainty in the strength of containment as a probability distribution. Some of the potential values of the containment strength that are

within the assessed range of uncertainty may be such that they are less than the bounding value of energy yield. Therefore, a probability of the strength being less than the energy yield could be assessed as the fraction of the distribution of possible containment strength values that is less than the energy yield. However, since this probability is a result of an assessment of epistemic uncertainty, it is itself an epistemic quantity. For that part of the distribution function on containment strength for which the strength is less than the bounding energy yield, the containment will always fail, and for the complementary part it will always survive. In this case the analysis has resulted in a pair of deterministic models of the phenomenon. In other words, if a core damage sequence of a certain class were to occur, it would always lead to one, but only one, of two (or in the general case more than two) possible outcomes.

The examples above were deliberately chosen to represent two different extreme approximations to representing a phenomenon. It has to be recognized that each is an approximation, and that both approximations have some elements of truth. The analyst must decide how to characterize the approach he is using, and that he feels most accurately represents his knowledge. In order to adopt the aleatory representation, however, the analyst must be able to construct, if not a mathematical, at least a mental model of the origins of the underlying variability and its consequences.

The situation is not always clear-cut. For example, consider the case of the assessment of the containment strength of plant X. It may be true that a large number of containments, built to the same specifications, may exhibit a variability in their strengths because of, among other things, for example, slight differences in concrete composition. On the other hand, there is no basis for considering that the strength of a particular containment is a variable from day to day. However, the variability among the strengths for the nominally identical plants could be used as a basis for characterizing the (epistemic) uncertainty in the containment strength of the specific plant X; the plant X could be regarded as a member of the population of plants with similarly designed containments. It could also be argued, however, that if, as suggested in Ref. 8, that the frequencies of sequences in a PRA can be interpreted as representing the results of a thought experiment in which the plant history is hypothetically observed a large number of times, the strength of the containment could be allowed to vary for each observation, which would lead to regarding the uncertainty on containment strength as an aleatory uncertainty. This would, however, be inconsistent with the treatment of other 'parameters' of the PRA model, such as the failure rate of the population of pumps in the plant. This is a parameter which is generally regarded as being different at different plants, because of differences in operating philosophy and maintenance practices for example. However, in the repetitions of the thought experiment it is treated as a constant. (The reader is warned that it is important not to confuse the Monte Carlo trials as repetitions of the thought experiment; the latter are performed implicitly and used to interpret the aleatory parameters such as unavailabilities, whereas the former are performed explicitly to represent the results obtained by using different values for the aleatory parameters.) There are clearly some issues for the analyst to address which require an interpretation of the entire PRA process. Some of these issues are subtle, and many analysts find themselves shying away from addressing them, making it difficult to distinguish between the two types of uncertainty. The next section illustrates why, however, it is necessary to try to do so.

5 THE NECESSITY OF SEPARATING ALEATORY AND EPISTEMIC UNCERTAINTY

As discussed in the previous section, because parameters that characterize aleatory uncertainty, or variability, and the probabilities that characterize epistemic uncertainty are dealing with different issues, the first being parameters of the model of the world[6] and the second uncertainty about what that model should look like, there is no option but to treat them separately. The fact that both these sets of parameters may obey the mathematical laws of probability, and thus can be called probabilities in the mathematical sense, does not alter that. That this is so should be clear from considering how new evidence can alter the PRA. By collecting more information we can indeed decrease our epistemic uncertainty with respect to parameter values and modelling issues, within the context of the structure of the model, using Bayes theorem as a basis. However, to decrease the aleatory content requires restructuring the model itself. This particular issue is not addressed in detail here. Instead, we illustrate the impact the distinction between the two types of characterizations of branch points has on the presentation of the results of a PRA, using a simple example.

Consider an event tree branch point that is representing the choice between two outcomes. Reference 7 illustrated this with a calculation related to the impact of steam explosion occurrence on consequence evaluation. As another example, suppose a core damage event tree has only two branches. The first (event A) represents the question of whether the required system is available or not. The probability associated with this branch is clearly a relative

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frequency: assume it has a value of 10^{-3}. The second event (event B) asks whether the system A will be effective in preventing core damage. Suppose there are two experts. The first expert recognizes that there are many unspecified variables that characterize the plant status and has determined that, for a certain subset of the range of those variables, system A will be effective, and for the complementary subset, it will not. Furthermore, he attempts to estimate the fraction of time that the plant is in the subset in which A will indeed be effective and comes up with a mean value of 0.99. In this case he will determine that the frequency of core damage is given by $f.(a + \bar{a}.b)$ where f is the frequency of the initiating event, assumed to be 10^{-1}/year, a is the probability of failure of event A, and b is the probability of failure of event B. In this case, expert 1's assessment of core damage frequency is approximately 10^{-1} $(10^{-3} + 0.999\ 1.10^{-2})$ or approximately 1.1×10^{-3}.

The second expert believes that the best way for him to describe his knowledge about event B is that he feels very strongly that system A is effective, but wants to express the fact that he is not certain, perhaps because he does not know the full range of possible conditions under which the system is to operate. Or, as another example, while it may be that the system has been designed to adequately address all foreseeable conditions, he may want to express a degree of doubt that the design process is flawless, or that the implementation of the design is perfect. He might calibrate the strength of this belief with historical data on the number of times there have been similar types of problems, but this does not represent an aleatory uncertainty for this application, for which the design is or is not adequate. In this case, he states that the fraction of times the yes branch is followed is 1 with probability p, and the fraction of times the no branch is followed is also 1 but with probability $1 - p$. That this is so is precisely because the PRA community has chosen to set up the PRA model of the world as one in which the experiment is repeated (hypothetically) many times, allowing the underlying boundary conditions to vary. In this way the PRA thought experiment is a sampling from the world of possible boundary conditions for the plant in question. When an analyst does not carry the variability in boundary conditions through, and characterizes certain branches as essentially being deterministic, they have to be understood as implying that all sequences go one way or the other, rather that some go one way, and some go another. Suppose that the second expert makes a statement that he is 99% confident that A will work, i.e., that the parameter p is 0.99. His assessment of core damage frequency therefore is $f.(a + \bar{a}.0)$, or 10^{-4} with 99% probability, or it is $f.(a + \bar{a}.1)$, or approximately 10^{-1} with 1% probability. This is a very different result from that of expert 1. The expected value of the core damage frequency, taken over the two hypotheses, will be the same as for expert 1. However, if the results were to be used to compare the calculated value with a safety goal, for example, the conclusions could be very different because of the different representations of uncertainty. Case 1 would give a unimodal distribution over core damage frequency whereas Case 2 would be bimodal and, as a result, the two experts would have very different levels of confidence in whether they meet the safety goal or not. Thus, making a distinction between the two representations is important to the interpretation of the results of the analysis.

It should be noted that, for expert 2's assessment to be meaningfully different from expert 1's assessment, he must use a different approach to assessing his probability, i.e., his must not be based on constructing a model of the underlying variability.

In all likelihood, expert 1 would also provide a statement of his uncertainty on his estimate for the likelihood of event B by constructing a probability distribution on the value of the likelihood. It is usually claimed that expert 2 should not provide an uncertainty about his probability. However, Mosleh & Bier[22] have pointed out that there are conditions under which it makes sense to do so.

6 CONCLUSIONS

When an analyst is trying to represent the impact of a variability in initial or boundary conditions that he cannot capture because of modelling or resources constraints, it has been customary to talk about his model of the world as being based on random processes, and the model will have parameters that characterize the system, or more accurately, the ensemble of 'identical' systems. Even if these values are assessed subjectively they are parameters of the model of the world and characterize aleatory uncertainty.

There is a significant difference in the impact on the results of an analysis between saying that both paths through a branch point are possible because of underlying variability in the boundary conditions, and saying that a branch point represents uncertainty as to which of two possibilities is the correct (and only) one. It is up to the subject matter expert for the particular modelling issue to determine the most appropriate way for him to characterize the issue. As discussed in Section 4 of this paper, it is clear that he must be very careful in formulating the problem and defining the event(s) of interest if his assessment is to be meaningful. If a PRA contains both types of approximations to the characterization of branch points, then as discussed in Section 5 of this paper, because of the impact on the interpretation of results,

the distinction between the two cases is important and should be maintained.

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the distinction between the two cases is important and should be maintained.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

June 20, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: PROPOSED RESOLUTION OF GENERIC SAFETY ISSUE-173A, "SPENT FUEL STORAGE POOL FOR OPERATING FACILITIES"

During the 473rd meeting of the Advisory Committee on Reactor Safeguards, June 7-9, 2000, we met with representatives of the NRC staff to discuss the proposed resolution of Generic Safety Issue (GSI)-173A, "Spent Fuel Storage Pool for Operating Facilities." We also had the benefit of the referenced documents.

Recommendations

1. The staff should defer closing out GSI-173A until the re-evaluation associated with spent fuel pool (SFP) accidents for decommissioning plants has been completed.

2. The staff should develop screening criteria for regulatory analyses that are appropriate for SFP accidents at operating reactors.

Discussion

The principal concerns of GSI-173A involve the potential for a sustained loss of SFP cooling capability and a potential for a substantial loss of SFP coolant inventory.

The staff had previously developed and implemented a generic spent fuel storage pool action plan to resolve concerns related to GSI-173A. This plan included plant-specific evaluations and regulatory analyses for safety enhancement backfits for plants that are more vulnerable to the GSI-173A concerns.

The staff has completed the review and evaluation of design features related to the SFP associated with each operating reactor. It found that existing structures, systems, and components related to storage of irradiated fuel provide adequate protection of public health and safety. Consequently,

the staff pursued regulatory analyses for safety enhancement backfits on a plant-specific basis. For these regulatory analyses, the staff used screening criteria for the frequency of "uncovery to within one foot of the top of fuel" or "loss of cooling for eight hours."

The screening criteria were:

$\leq 10^{-6}$/yr	No action justified
10^{-6}/yr to 10^{-5}/yr	Further evaluation needed
$\geq 10^{-5}$/yr	Proceed to value-impact evaluation

With this choice of screening criteria, the staff determined that no further regulatory actions were warranted.

The screening criteria, which constituted the primary basis for the staff's findings, are essentially equivalent to the criteria in the Regulatory Analysis Guidelines. The criteria in the Regulatory Analysis Guidelines are derived from the prompt fatality quantitative health objective (QHO) of the Safety Goal Policy Statement. These are appropriate surrogates for this QHO for reactor accident source terms (fission product releases) driven by steam-zircaloy oxidation. As noted in our report of April 13, 2000, which is related to SFP accident risk at decommissioning nuclear power plants, it is very likely that the source terms for SFP accidents will be significantly different from those for operating reactor accidents. The fission product release from spent fuel accidents is most likely driven by air oxidation of the zircaloy clad. Under such circumstances, there is convincing evidence that there may be substantial release of the ruthenium inventory as the volatile oxide, as well as release of significant quantities of "fuel fines" through a decrepitation process.

Such differences in source terms have significant implications. Ruthenium has relatively long half-life isotopes, its inventory in spent fuel is substantial, and its biological consequences are severe. In connection with decommissioning plants, the staff estimated that prompt fatalities due to an SFP fire could increase by as much as two orders of magnitude if the source term is assumed to include 100-percent release of ruthenium compared to essentially zero release. In addition, the societal dose could double and the cancer fatalities could increase four-fold for this estimated source term. The consequences of actinide releases associated with either fuel decrepitation or matrix-stripping have not yet been evaluated. With emergency response measures, the limiting consideration might well no longer be prompt fatalities. The staff should assess the impact of the different source term on latent fatalities and land contamination.

Because of these differences in the source term, the screening criteria used in this application appear to be inappropriate as surrogates for the prompt fatality QHO related to SFP accidents at operating reactors. A proper surrogate could lead to changes in the conclusions that the staff has reached.

Before closing out GSI-173A and developing the Standard Review Plan and regulatory guidance, the staff should await the results of the proposed re-evaluation of SFP accidents for decommissioning plants and should re-evaluate the regulatory analysis screening criteria for application to SFP accidents at operating reactors.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

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Thompson Rept. Exh.: Powers, 2000b

ACRSR-1885

PDR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

April 13, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: DRAFT FINAL TECHNICAL STUDY OF SPENT FUEL POOL ACCIDENT RISK AT DECOMMISSIONING NUCLEAR POWER PLANTS

During the 471st meeting of the Advisory Committee on Reactor Safeguards, April 5-7, 2000, we met with representatives of the NRC staff and discussed the subject document. We also had the benefit of the documents referenced, which include the available stakeholders comments. This report is in response to the Commission's request in the Staff Requirements Memorandum dated December 21, 1999, that the ACRS perform a technical review of the validity of the draft study and risk objectives.

BACKGROUND

Decommissioning plants are subject to many of the same regulatory requirements as operating nuclear plants. Because of the expectation that the risk will be lower at decommissioning plants, particularly as time progresses to allow additional decay of fission products, some of these requirements may be inappropriate. Exemptions from the regulations are frequently requested by licensees after a nuclear power plant is permanently shut down. To increase the efficiency and effectiveness of decommissioning regulations, the staff has engaged in rulemaking activities that would reduce the need to routinely process exemptions. The staff has undertaken the technical study and risk analysis discussed here to provide a firm technical basis for rulemaking concerning several exemption issues.

In the draft study the staff has concluded that, provided certain industry decommissioning commitments are implemented at the plants, after one year of decay time the risk associated with spent fuel pool fires is sufficiently low that emergency planning requirements can be significantly reduced. It also concluded that after five years the risk of zirconium fires is negligible even if the fuel is uncovered and that requirements intended to ensure spent fuel cooling can be reduced.

RECOMMENDATIONS

1. The integrated rulemaking on decommissioning should be put on hold until the staff provides technical justification for the proposed acceptance criterion for fuel uncovery frequency. In particular, the staff needs to incorporate the effects of enhanced release of ruthenium under air-oxidation conditions and the impact of the MELCOR Accident

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Consequence Code System (MACCS) code assumptions on plume-related parameters in view of the results of expert elicitation.

2. The technical basis underlying the zirconium-air interactions and the criteria for ignition needs to be strengthened. In particular, the potential impact of zirconium-hydrides in high burnup fuel and the susceptibility of the clad to breakaway oxidation need to be addressed.

3. Uncertainties in the risk assessment need to be quantified and made part of the decisionmaking process.

DISCUSSION

The staff's conclusion that the risk after one year of decay time is sufficiently low that emergency planning requirements can be reduced is based partially on the assessed value of fuel uncovery frequency (3.4 x 10^{-6}/yr) being less than the Regulatory Guide 1.174 large, early release frequency (LERF) acceptance value (1x10^{-5}/yr). This LERF risk-acceptance value was derived to be a surrogate for the Safety Goal early fatality quantitative health objectives (QHO) *for operating reactors*. The derivation from the QHO is based, however, on the fission product releases that occur under severe accident conditions which are driven by steam oxidation of the zircaloy and the fuel. These releases include only insignificant amounts of ruthenium. Under air-oxidation conditions of spent fuel fires, significant data indicate much enhanced releases of ruthenium as the very volatile oxide. Indications are that, under air oxidation conditions, the release fractions of ruthenium may be equivalent to those for iodine and cesium. In the accident at Chernobyl significant releases of ruthenium were observed and attributed to the interactions of fuel with air.

These findings have significant implications. The ruthenium inventory in spent fuel is substantial. Ruthenium has a biological effectiveness equivalent to that of iodine-131 and has a relatively long half-life. If there are significant releases of ruthenium, the Regulatory Guide 1.174 LERF value may not be an appropriate surrogate for the prompt fatality QHO. In addition, because of the relatively long half-life of ruthenium-106, it is likely that the early fatality QHO would no longer be the controlling consequence.

In response to our concerns about the effects of substantial ruthenium release, the staff has made additional MACCS calculations in which it assumed 100 percent release of the ruthenium inventory. For a one-year decay time with no evacuation, the prompt fatalities increased by two orders of magnitude over those in the report which did not include ruthenium release, the societal dose doubled and the cancer fatalities increased four-fold.

Our concern is not just with ruthenium. We are concerned with the appropriateness of the entire source term used in the study. There is a known tendency for uranium dioxide in air to decrepitate into fine particles. The decrepitation is caused by lattice strains produced as the dioxide reacts to form U_3O_8. This decrepitation is a bane of thermogravimetric studies of air oxidation of uranium dioxide since it can cause fine particles to be entrained in the flowing air of the apparatus. This suggests that decrepitating fuel would be readily entrained in vigorous natural convection flows produced in an accident at a spent fuel pool. The decrepitation process provides a low-temperature, mechanical, release mechanism for even very refractory

radionuclides. The staff did consider the possibility that "fuel fines" could be released from fuel with ruptured cladding. It did not, however, believe these fuel fines could escape the plant site. Nevertheless, the staff considered the effect of a $6x10^{-6}$ release fraction of fines. This minuscule release fraction did not significantly affect the calculated findings. There is no reason to think that such a low release fraction would be encountered with decrepitating fuel.

Consequences of accidents involving a spent fuel pool were analyzed using the MACCS code. The staff has completed an expert opinion elicitation regarding the uncertainties associated with many of the critical features of the MACCS code. The findings of this elicitation seem not to have been considered in the analyses of the spent fuel pool accident. One of the uncertainties in MACCS identified by the experts is associated with the spread of the radioactive plume from a power plant site. The spread expected by the experts is much larger than what is taken as the default spread in the MACCS calculations. There is no indication that the staff took this finding into account in preparing the consequence analyses. In addition, the initial plume energy assumed in the MACCS calculations, which determines the extent of plume rise, was taken to be the same as that of a reactor accident rather than one appropriate for a zirconium fire. We suspect, therefore, that the consequences found by the staff tend to overestimate prompt fatalities and underestimate land contamination and latent fatalities just because of the narrow plume used in the MACCS calculations and the assumed default plume energy.

The staff needs to review the air oxidation fission products release data from Oak Ridge National Laboratory and from Canada that found large releases of cesium, tellurium, and ruthenium at temperatures lower than 1000 °C. Based on these release values for ruthenium, and incorporating uncertainties in the MACCS plume dispersal models, the consequence analyses should be redone.

Based on the results of this reevaluation of the consequences, the staff should determine an appropriate LERF for spent fuel fires that properly reflects the prompt fatality QHO and the potential for land contamination and latent fatalities associated with spent fuel pool fires.

In developing risk-acceptance criteria associated with spent fuel fires, the staff should also keep in mind such factors as the relatively small number of decommissioning plants to be expected at any given time and the short time at which they are vulnerable to a spent fuel pool fire.

We also have difficulties with the analysis performed to determine the time at which the risk of zirconium fires becomes negligible. In previous interactions with the staff on this study, we indicated that there were issues associated with the formation of zirconium-hydride precipitates in the cladding of fuel especially when that fuel has been taken to high burnups. Many metal hydrides are spontaneously combustible in air. Spontaneous combustion of zirconium-hydrides would render moot the issue of "ignition" temperature that is the focus of the staff analysis of air interactions with exposed cladding. The staff has neglected the issue of hydrides and suggested that uncertainties in the critical decay heat times and the critical temperatures can be found by sensitivity analyses. Sensitivity analyses with models lacking essential physics and chemistry would be of little use in determining the real uncertainties.

The staff analysis of the interaction of air with cladding has relied on relatively geriatric work. Much more is known now about air interactions with cladding. This greater knowledge has come in no small part from studies being performed as part of a cooperative international

program (PHEBUS FP) in which NRC is a partner. Among the findings of this work is that nitrogen from air depleted of oxygen will interact exothermically with zircaloy cladding. The reaction of zirconium with nitrogen is exothermic by about 86,000 calories per mole of zirconium reacted. Because the heat required to raise zirconium from room temperature to melting is only about 18,000 calories per mole, the reaction enthalpy with nitrogen is ample. In air-starved conditions, the reaction of air with zirconium produces a duplex film in which the outer layer is zirconium dioxide (ZrO_2) and the inner layer is the crystallographically different compound zirconium nitride (ZrN). The microscopic strains within this duplex layer can lead to exfoliation of the protective oxide layer and reaction rates that deviate from parabolic rates. These findings may well explain the well-known tendency for zirconium to undergo breakaway oxidation in air whereas no such tendency is encountered in either steam or in pure oxygen. Because of these findings, we do not accept the staff's claim that it has performed "bounding" calculations of the heatup of Zircaloy clad fuel even when it neglects heat losses.

The staff focuses its analysis of the reactions of gases with fuel cladding on a quantity they call an "ignition temperature." The claim is that this is the temperature of self-sustained reaction of gas with the clad. Gases will react with the cladding at all temperatures. In fact, at temperatures well below the "conservative ignition temperature" identified by the staff, air and oxygen will react with the cladding quite smoothly and at rates sufficient to measure. Data in these temperature ranges well below the "ignition" temperature form much of the basis for the correlations of parabolic reaction rates with temperature. We believe that the staff should look for a condition such that the increase with temperature of the heat liberation rate by the reaction of gas with the clad exceeds the increase with temperature of the rate of heat losses by radiation and convection. Finding this condition requires that there be high quality analyses of the heat losses and that the heat of reaction be properly calculated. Since staff has neglected any reaction with nitrogen and did not consider breakaway oxidation (causes for the deviations from parabolic reaction rates), it has not made an appropriate analysis to find this "ignition temperature."

In fact, the search for the ignition temperature may be the wrong criterion for the analysis. The staff should also be looking for the point at which cladding ruptures and fission products can be released. Some fraction of the cladding may be ruptured before any exposure of the fuel to air occurs. Even discounting this, one still arrives at much lower temperature criteria for concern over the possible release of radionuclides.

There are other flaws in the material interactions analyses performed as part of the study. For instance, in examining the effects of aluminum melting, the staff seems to not recognize that there is a very exothermic intermetallic reaction between molten aluminum and stainless steel. Compound formation in the Al-Zr system suggests a strong intermetallic reaction of molten aluminum with fuel cladding as well. The staff focuses on eutectic formations when, in fact, intermetallic reactions are more germane to the issues at hand.

We are concerned about the conservative treatment of seismic issues. Risk-informed decisionmaking regarding the spent fuel pool fire issues should use realistic analysis, including an uncertainty assessment.

Because the accident analysis is dominated by sequences involving human errors and seismic events which involve large uncertainties, the absence of an uncertainty analysis of the

frequencies of accidents is unacceptable. The study is inadequate until there is a defensible uncertainty analysis.

The risk posed by fuel uncovery in spent fuel pools for decommissioning plants may indeed be low, however, the technical shortcomings of this study are significant and sufficient for us to recommend that rulemaking be put on hold until the inadequacies discussed herein are addressed by the staff.

Sincerely

Dana A. Powers

Dana A. Powers
Chairman

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3. Memorandum dated December 21, 1999, from Anette L. Vietti-Cook, Secretary of the Commission, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements -SECY-99-168 - Improving Decommissioning Regulations for Nuclear Power Plants.
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5. Letter dated December 16, 1999, from William D. Travers, Executive Director for Operations, NRC, to Dana A. Powers, Chairman, ACRS, Subject: Spent Fuel Fires Associated With Decommissioning.
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Thompson Rept. Exh.: NRC, 1975
(excerpts)

WASH-1400
(NUREG 75/014)

CALCULATION OF REACTOR ACCIDENT CONSEQUENCES

APPENDIX VI
to
REACTOR SAFETY STUDY

U.S. NUCLEAR REGULATORY COMMISSION
OCTOBER 1975

TABLE VI 3-1 INITIAL ACTIVITY OF RADIONUCLIDES IN THE NUCLEAR REACTOR CORE AT THE TIME OF THE HYPOTHETICAL ACCIDENT

No.	Radionuclide	Radioactive Inventory Source (curies x 10^8)	Half-Life (days)
1	Cobalt-58	0.0078	71.0
2	Cobalt-60	0.0029	1,920
3	Krypton-85	0.0056	3,950
4	Krypton-85m	0.24	0.183
5	Krypton-87	0.47	0.0528
6	Krypton-88	0.68	0.117
7	Rubidium-86	0.00026	18.7
8	Strontium-89	0.94	52.1
9	Strontium-90	0.037	11,030
10	Strontium-91	1.1	0.403
11	Yttrium-90	0.039	2.67
12	Yttrium-91	1.2	59.0
13	Zirconium-95	1.5	65.2
14	Zirconium-97	1.5	0.71
15	Niobium-95	1.5	35.0
16	Molybdenum-99	1.6	2.8
17	Technetium-99m	1.4	0.25
18	Ruthenium-103	1.1	39.5
19	Ruthenium-105	0.72	0.185
20	Ruthenium-106	0.25	366
21	Rhodium-105	0.49	1.50
22	Tellurium-127	0.059	0.391
23	Tellurium-127m	0.011	109
24	Tellurium-129	0.31	0.048
25	Tellurium-129m	0.053	0.340
26	Tellurium-131m	0.13	1.25
27	Tellurium-132	1.2	3.25
28	Antimony-127	0.061	3.88
29	Antimony-129	0.33	0.179
30	Iodine-131	0.85	8.05
31	Iodine-132	1.2	0.0958
32	Iodine-133	1.7	0.875
33	Iodine-134	1.9	0.0366
34	Iodine-135	1.5	0.280
35	Xenon-133	1.7	5.28
36	Xenon-135	0.34	0.384
37	Cesium-134	0.075	750
38	Cesium-136	0.030	13.0
39	Cesium-137	0.047	11,000
40	Barium-140	1.6	12.8
41	Lanthanum-140	1.6	1.67
42	Cerium-141	1.5	32.3
43	Cerium-143	1.3	1.38
44	Cerium-144	0.85	284
45	Praseodymium-143	1.3	13.7
46	Neodymium-147	0.60	11.1
47	Neptunium-239	16.4	2.35
48	Plutonium-238	0.00057	32,500
49	Plutonium-239	0.00021	8.9×10^6
50	Plutonium-240	0.00021	2.4×10^6
51	Plutonium-241	0.034	5,350
52	Americium-241	0.000017	1.5×10^5
53	Curium-242	0.0050	163
54	Curium-244	0.00023	6,630

TABLE VI C-2 PHOTON DOSE-CONVERSION FACTORS FOR EXPOSURE TO CONTAMINATED GROUND (rem/Ci-m^2) (TIME INTEGRAL DOSE TO N DAYS)[(a)]

Radionuclide	Whole Body[(b)]		Total Marrow		Lung		Testes	
	1 Day	7 Days	1 Day	7 Days	1 Day	7 Days	1 Day	7 Days
CO-58	3.29E+02	2.74E+03	3.67E+02	2.50E+03	3.98E+02	2.19E+03	3.08E+02	2.09E+03
CO-60	8.48E+02	5.85E+03	8.89E+02	6.22E+03	7.39E+02	5.58E+03	6.91E+02	4.80E+03
KR-85	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
KR-85M	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
KR-87	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
KR-88	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
RB-86	2.93E+01	1.85E+02	3.22E+01	2.02E+02	2.75E+01	1.73E+02	2.66E+01	1.66E+02
SR-89	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
SR-90	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
SR-91	1.68E+02	2.05E+02	1.96E+02	2.39E+02	1.59E+02	1.93E+02	1.70E+02	2.07E+02
Y-90	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Y-91	8.82E-01	5.91E+00	8.98E-01	6.96E+00	8.39E-01	5.66E+00	6.61E-01	4.44E+00
ZR-95	2.47E+02	1.77E+03	2.85E+02	2.04E+03	2.32E+02	1.67E+03	2.46E+02	1.76E+03
ZR-97	3.33E+02	5.38E+02	4.02E+02	6.52E+02	3.14E+02	5.10E+02	3.50E+02	5.59E+02
NB-95	2.49E+02	1.54E+03	2.73E+02	1.39E+03	2.33E+02	1.56E+03	2.25E+02	1.49E+03
MO-99	7.73E+01	3.25E+02	1.07E+02	4.65E+02	6.95E+01	2.91E+02	7.05E+01	3.32E+02
TC-99M	1.52E+01	1.62E+01	2.70E+01	2.88E+01	1.25E+01	1.35E+01	1.41E+01	1.51E+01
RU-103	1.75E+02	1.16E+03	2.14E+02	1.42E+03	1.65E+02	1.09E+03	1.95E+02	1.30E+03
RU-105	7.23E+01	7.94E+01	9.04E+01	9.99E+01	6.77E+01	7.37E+01	8.93E+01	8.91E+01
RU-106	6.51E+01	4.56E+02	7.96E+01	5.54E+02	6.17E+01	4.59E+02	7.15E+01	4.97E+02
RH-105	2.29E+01	5.67E+01	3.32E+01	8.55E+01	1.95E+01	5.01E+01	2.07E+01	7.37E+01
TE-127	6.77E-01	8.13E-01	8.43E-01	1.01E+00	6.38E-01	7.67E-01	7.58E-01	9.14E-01
TE-127M	7.99E+00	5.84E+01	1.27E+01	9.20E+01	4.40E+00	3.37E+01	1.07E+01	7.77E+01
TE-129	1.98E+00	1.98E+00	2.44E+00	2.44E+00	1.83E+00	1.83E+00	2.17E+00	2.17E+00
TE-129M	3.63E+01	2.46E+02	4.54E+01	3.97E+02	3.28E+01	2.22E+02	4.04E+01	2.74E+02
TE-131M	3.01E+02	9.60E+02	4.36E+02	1.10E+03	3.56E+02	8.90E+02	3.47E+02	8.86E+02
TE-132	6.77E+02	3.08E+03	8.04E+02	3.53E+03	6.35E+02	2.98E+03	6.07E+02	3.08E+03
SB-127	2.12E+02	9.20E+02	2.57E+02	1.11E+03	2.90E+02	8.65E+02	2.29E+02	9.35E+02
SB-129	1.00E+02	1.04E+02	1.12E+02	1.16E+02	9.47E+01	9.78E+01	9.25E+01	9.57E+01
I-131	1.28E+02	7.08E+02	1.59E+02	8.73E+02	1.21E+02	6.53E+02	1.43E+02	7.85E+02
I-132	1.07E+02	1.07E+02	1.23E+02	1.23E+02	1.01E+02	1.01E+02	1.05E+02	1.05E+02
I-133	1.63E+02	3.11E+02	1.93E+02	3.75E+02	1.54E+02	2.91E+02	1.70E+02	3.22E+02
I-134	4.14E+01	4.14E+01	4.56E+01	4.56E+01	3.88E+01	3.88E+01	3.76E+01	3.76E+01
I-135	2.52E+02	2.85E+02	2.77E+02	3.18E+02	2.38E+02	2.69E+02	2.18E+02	2.52E+02
XE-133	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
XE-135	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
CS-134	5.30E+02	3.69E+03	6.10E+02	4.26E+03	4.97E+02	3.47E+03	5.27E+02	3.87E+03
CS-136	6.04E+02	4.10E+03	7.79E+02	4.68E+03	6.37E+02	3.32E+03	6.33E+02	3.92E+03
CS-137	1.86E+02	1.31E+03	2.28E+02	1.60E+03	1.76E+02	1.24E+03	2.08E+02	1.46E+03
BA-140	2.13E+02	3.65E+03	2.42E+02	3.90E+03	2.02E+02	3.46E+03	1.93E+02	3.11E+03
LA-140	6.49E+02	1.80E+03	6.88E+02	1.92E+03	6.12E+02	1.71E+03	5.20E+02	1.56E+03
CE-141	2.77E+01	1.52E+02	4.93E+01	3.24E+02	2.28E+01	1.50E+02	2.63E+01	1.73E+02
CE-143	9.00E+01	2.24E+02	1.24E+02	3.90E+02	3.06E+01	2.79E+02	1.05E+02	2.59E+02
CE-144	1.72E+01	1.29E+02	2.39E+01	1.67E+02	1.52E+01	1.97E+02	1.46E+01	1.03E+02
PR-143	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
ND-147	5.20E+01	3.05E+02	7.28E+01	4.26E+02	4.63E+01	2.70E+02	5.71E+01	3.34E+02
NP-239	5.89E+01	2.02E+02	9.53E+01	3.26E+02	5.09E+01	1.74E+02	6.07E+01	2.35E+02
PU-238	8.95E-01	6.20E+00	7.17E-01	5.02E+00	1.62E-01	1.14E+00	7.19E-01	4.97E+00
PU-239	3.76E-01	2.63E+00	3.56E-01	2.49E+00	8.88E-02	6.22E-01	3.18E-01	2.23E+00
PU-240	7.84E-01	5.47E+00	6.53E-01	4.57E+00	1.55E-01	1.08E+00	6.25E-01	4.38E+00
PU-241	4.52E-05	2.21E-03	9.23E-05	4.52E-03	3.13E-05	1.56E-03	3.95E-05	1.94E-03
AM-241	2.96E+01	1.43E+02	4.21E+01	2.95E+02	1.45E+01	1.02E+02	1.20E+01	1.26E+02
CM-242	7.92E-01	5.46E+00	6.14E-01	4.25E+00	1.31E-01	9.05E-01	6.29E-01	4.35E+00
CM-244	4.96E+00	3.46E+01	9.73E+00	6.81E+01	3.73E+00	2.61E+01	4.40E+00	3.98E+01

(a) Note: Noble gases do not deposit on the ground.

(b) For organs not listed the dose-conversion factors are similar to those of the whole body.

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NRC Report on the January 25, 1982 Steam Generator Tube Rupture at R. E. Ginna Nuclear Power Plant

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significant thermal shock to the reactor vessel wall during this event has not been ruled out.

1.3.2 Human Factors Considerations

The NRC Task Force conducted a human factors review of the Control Room and Technical Support Center designs, a review of personnel responses, and a review of the procedures as they supported the steam generator tube rupture event. The review also included a comparison of the procedures with the Westinghouse Owners' Group guidelines. These reviews were accomplished through interviews with plant personnel, examination of the Control Room, the Technical Support Center, the procedures as they compared with human factors guidelines, and partial walk-throughs of the event with operators.

The Control Room and Technical Support Center physical facilities were satisfactory to support the activities required to mitigate the consequences of this steam generator tube rupture event. In addition, the response of the licensee's plant staff to the event was good.

In general, based on the training and experience of the plant staff, the applicable procedures were adequate for coping with the event. Problems with the procedures are identified.

1.3.3 Radiological Consequences

The NRC Task Force estimated the curies available for release from the reactor coolant system, the amount of activity transferred to the faulted steam generator, and the activity released to the environment as a function of time. Both airborne and liquid releases were estimated. Airborne release figures were then converted to projected offsite dose figures using conservative dispersion models based on existing weather conditions.

On- and offsite radionuclide release and exposure measuring devices were read and results analyzed. Risk to the public and licensee personnel were then estimated.

Most of the radionuclides released from Ginna were released during the first 3 hours of the event. During this period the wind was blowing toward the southeast. Snow and the moist cold air caused a large fraction of the radioiodines and particulates released from Ginna to deposit on the Ginna site, rather than to remain airborne beyond the site boundary. The Task Force estimated that airborne releases to an owner-controlled, unrestricted area exceeded the limits in 10 CFR 20. Other offsite releases were estimated to be less than 25% of the limit for unrestricted areas. All releases would result in doses which were significantly less than the 10 CFR 100 guidelines.

Potential health impacts from the estimated doses and predicted exposures were insignificant compared with the natural incidence of cancer fatalities and genetic abnormalities.

1.3.4 Institutional Response

Various organizations, including the licensee, State and local governments, NRC, and other Federal agencies responded to the event at Ginna.

The licensee had primary responsibility for resolving the conditions that existed at the plant. Prescribed initial notifications by the licensee to the State, local counties, and NRC were completed very early in the event, and interactions throughout the event were maintained among all the participants.

The Nuclear Regulatory Commission, using the resources of the Senior Resident Inspector, the Region I Base and Site Teams, and the Headquarters Executive and Analytical Teams, monitored the licensee's actions in response to the event to assure that these actions were correct and appropriate.

The State of New York and Wayne and Monroe Counties were promptly notified by the licensee. They responded by activating their Emergency Operations Centers and sending representatives to the site. Monroe County also fielded offsite radiological monitoring teams and reported results back to the Emergency Operations Center throughout the day. Twice during the first day of the event,

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Probabilistic Risk Assessment (PRA) Reference Document

Final Report

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Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

EXECUTIVE SUMMARY

This is an executive summary of the important conclusions of the report. This summary is in the form of listings of the more important findings. Since there are important exceptions and nuances difficult to portray in such a summary, the reader is strongly urged to read both the individual chapters for more detailed findings and supporting rationale, and the appendixes for a fuller understanding of the technical bases.

What Is PRA?

PRA is an analysis that: (1) identifies and delineates the combinations of events that, if they occur, will lead to a severe accident (i.e., severe core damage or core melt) or any other undesired event; (2) estimates the frequency of occurrence for each combination; and (3) estimates the consequences. As practiced in the field of nuclear power, PRAs focus on core-melt accidents, since they pose the greatest potential risk to the public. The PRA integrates into a uniform methodology the relevant information about plant design, operating practices, operating history, component reliability, human actions the physical progression of core-melt accidents, and potential environmental and health effects, usually in as realistic a manner as possible.

What Is The State of Development of PRA?

- Qualitative systems analysis (logic modeling) for internal accident initiators has reached a relatively high level of development, where development is defined as the degree of confidence that changes in the state of knowledge will not result in substantial changes in the major insights drawn from PRAs. Therefore, a relatively high degree of confidence can be placed in the qualitative insights drawn with regard to dominant accident sequences from internal events and their more important contributors. One area where improvement is needed is the modeling of common-cause failures.

- Qualitative systems analysis for external accident initiators (seismic, fire, flood) has reached a medium level of development, which means ~~that~~ a fair degree of confidence can be placed in the qualitative insights drawn with regard to dominant accident sequences from external events and their more important contributors. Again, the modeling of common-cause failures needs to be improved for all initiators.

- Advances have been made in the modeling of human performance, and the likelhood of operator errors generally can be quantified to order-of-magnitude precision, particularly those errors which arise from failure to follow written procedures. However, the quantification of errors of misdiagnosis and potential recovery actions to terminate an accident sequence has substantial uncertainty and needs improvement.

- The data base is fairly good for events of high frequency,* but poor for events of low frequency, such as failures of very reliable systems (e.g., the reactor protection system), the occurrence of high-magnitude seismic events, or the occurrence of common-cause failures. This means that internally initiated accidents normally can be quantified with a fair degree of confidence, but normally one has only poor confidence in the quantification of externally initiated accidents because the results tend to be dominated by low-frequency initiators. It is not likely that the data base for low-frequency events will improve appreciably in the near future.

- Estimates of source terms are currently made with poor confidence, principally because of lack of knowledge regarding the phenomena of core-melt progression, radionuclide transport inside the reactor coolant system and the containment, and containment performance. Extensive research is under way which should result in substantially improving the state of knowledge of the phenomenology of core melt, radionuclide transport, and the resultant containment loadings and response. However, uncertainties will likely remain quite large.

- The calculation of consequences, given a source term and the meteorology, can be performed with reasonably high confidence. However, there is still a stochastic uncertainty associated with the actual meteorology at the time of a major radiological release, which means that the actual consequences as a function of location away from the site cannot be predicted with much precision prior to an accident. Also, the actual behavior of the affected population during emergency actions (sheltering, evacuation) is not well understood.

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*As used herein, high-frequency events are those which are often observed in plant operation. Low-frequency events are those rarely observed, having a return frequency less than once in 1000 reactor-years.

What Are The Principal Conclusions Regarding Uncertainties?

- Uncertainties in estimating core-melt frequency due to internal initiators are generally reported to be an order of magnitude or less above and below the best estimate. However, these estimates may not include the effects of modeling assumptions.

- Uncertainties in estimating core-melt frequency due to external initiators currently are generally about a factor of 10 to 30 above and below the best estimate.

- Uncertainties in estimates of the source term presently are very large but have not been well analyzed in PRAs.

- Uncertainties in mean early fatalities, given a large source term, could range from about a factor of 5 above the best estimate to nearly zero, in large part due to assumptions made about emergency actions taken, including evacuation.

- Uncertainties in mean population dose, given a source term, would lie within a factor of 3 or 4 of the best estimate, while uncertainties in estimates of latent cancer deaths could be approximately a factor of 10 above and below the best estimate.

- There is some question whether the statistical techniques employed in PRAs have been implemented properly, particularly in assigning probability distributions to parameters based on limited data.

- Completeness does not seem to be the principal limitation when examining the general insights gained from a PRA on dominant sequences, since the data base is large enough so that a rare and unusual type of failure likely would not affect the conclusions regarding dominant sequences. However, from time to time some issues (e.g., pressurized thermal shock) will warrant regulatory attention even though they had not previously been considered important from either a probabilistic or a deterministic perspective.

- Design and construction errors should already be part of the data base for higher-frequency events and thus would be inherently included in a PRA. However, such errors for low-frequency events probably would not be in the data base. It is unclear what uncertainties this would imply for the PRA estimates.

- PRAs could be made more reproducible from one analyst to the next by specifying the data, modeling, success/failure assumptions, and phenomenology to be used. However, even under such circumstances, differences of a factor of 3 or more between analysts in estimates of core-melt frequency would not be surprising.

- One method for propagating data uncertainties (the Bayesian approach) is reasonably well developed. Approaches based on classical statistics need to be explored. More work needs to be done on propagating knowledge uncertainties (e.g., phenomena), and uncertainty and sensitivity analyses need to be more widely used and better organized and displayed to assure that users of PRA information are better informed as to the important uncertainties.

<u>To What Extent Have PRA Results Been Validated?</u>

- The frequency estimated for severe core-damage accidents is usually low (on the order of once in 10,000 reactor-years). It is not possible to validate the results directly because sufficient data does not exist. Therefore, it is necessary to attempt to validate as many of the constituent parts of the PRA as possible.

- Plant-specific design or operational features can have an important influence on dominant accident sequences; therefore, a generic validation of results is difficult.

- Estimates of accident-initiator frequency are reasonably well validated by plant data for those events which occur relatively often.

- To some extent, failure-rate estimates have been validated, particularly for active components.

- Some validation of computer codes has occurred, mainly through benchmark comparisons. Much remains to be accomplished in this area.

- The validation level of a PRA is not thorough or detailed; however, this level of validation is usually not much worse than the degree of validation achieved by alternative analytical tools.

<u>Does Operating Experience Reasonably Conform To The Results of PRAs?</u>

- Transient information and failure data are used as input to the PRAs. Transient information is reasonably reliable; however, the data base for equipment and human failures needs improvement.

- The initial results of the accident precursor program being conducted by the Office of Nuclear Regulatory Research, NRC, indicate a fair degree of agreement (order of magnitude) with PRA results relative to the estimated likelihood of core melt as well as to the major accident contributors.

Can Generic Insights Be Drawn From PRAs?

- Generic insights can be drawn from PRA with regard to aspects of design and operations important to the dominant accident sequences. However, plant-specific features could be of significant importance to the estimation of core-melt frequency or risk.

- The degree to which generic insights can be relied upon in regulation depends on the regulatory use and the specific safety issue under consideration.

What Are The Major Insights That Have Been Drawn From Present PRAs?

Note: Only global insights are provided below. The reader is referred to Chapter 3 and Appendix B for more detailed insights.

- The process of performing PRA studies yields extremely valuable engineering and operational insights regarding the integrated safety performance of nuclear power plants.

- The estimated frequency of core melt is higher than had been thought prior to performing the Reactor Safety Study; however, most core melts are not expected to result in large offsite radiological consequences.

- The range of core-melt frequency point estimates in U.S. PRAs published to date covers about two orders of magnitude (about 10^{-5} to 10^{-3} per reactor-year). It is extremely difficult to pinpoint generic reasons for the difference.

- The specific features of dominant accident sequences and the estimates of risk vary significantly from plant to plant, even though plants meet all applicable NRC regulatory requirements.

- Estimates of early fatalities and injuries are very sensitive to source-term magnitudes, and a major factor in the estimate of source-term magnitude is the timing of containment failure (early or late compared to core melt). With large source terms, they are sensitive to emergency response assumptions, but this dependence decreases in importance if source terms are reduced.

- Estimates of latent cancer fatalities are sensitive to source-term magnitudes, but site-to-site differences are relatively small for a given source term.
- Estimated onsite economic losses resulting from a core-melt accident are generally much larger than estimated offsite economic losses.
- Generally, airborne radiological pathways are much more important to risk than liquid pathways.
- Accidents beyond the design basis (such as those caused by earthquakes more severe than the safe-shutdown earthquake) are the principal contributors to public risk.
- Small LOCAs and transients are usually dominant contributors to estimated core-melt frequency and risk, while large LOCAs usually are not.
- Dominant contributors to risk are not necessarily the same accident sequences as the dominant contributors to core-melt frequency.
- Human interactions, including test and maintenance considerations, are extremely important contributors to the safety of plants.
- Common-cause (dependent) failures are important contributors to estimates of core-melt frequency and plant risk.
- Earthquakes, internal fires, and floods seem to play an important role in estimates of core-melt frequency and plant risk, although this tentative conclusion appears to be highly plant specific.
- The failure of long-term decay heat removal is a major functional contributor to estimated core-melt frequency.
- The reliability of systems, components, and human actions important to safety must be maintained during operation. Degradation in their reliability can sharply increase risk or the likelihood of core melt.

<u>What Is The Usefulness of PRA In The Regulation of Nuclear Power Plants?</u>

- PRA results are useful, provided that more weight is given to the qualitative and relative insights regarding design and operations, rather than the precise absolute magnitude of the numbers generated.

- It must be remembered that most of the uncertainties associated with an issue are inherent to the issue itself rather than artifacts of the PRA analysis. The PRA does tend to identify and highlight these uncertainties, however.

- PRA results have useful application in the prioritization of regulatory activities, development of generic regulatory positions on potential safety issues, and the assessment of plant-specific issues. The degree of usefulness depends on the regulatory application as well as the nature of the specific issue, and the reader is referred to Chapter 2 for more detail and specific examples.

- PRAs are not very useful from a quantitative standpoint for some issues. However, PRAs can still provide useful regulatory insights even for these issues. For example, the risk from sabotage is difficult to quantify due to uncertainty in the frequency of attempted acts and the nature of and likelihood of success for sabotage attempts; however, PRA methods can still provide good qualitative insights with regard to important (vital) plant areas and weaknesses.

- The need for plant-specific PRAs depends on the intended application. Most regulatory uses would not be dependent on the availability of a plant-specific PRA.

- The basic attributes of a PRA are not highly compatible with a safety-goal structure that would require strict numerical compliance on the basis of the quantitative best estimates of a PRA. However, there could be useful application if the structure were less strict or the goals were set so conservatively that there would be little regulatory concern if the actual value substantially exceeded those goals.

- The results of a PRA should only be one consideration in regulatory decisions, i.e., they should not replace other conventional considerations. When assessing the weight to be given to PRAs in a decision, one should consider:

  - The scope and depth of the PRA (i.e., does the nature of the PRA reasonably match the needs of the decision);

  - The degree of realism embodied in the PRA;

  - The results of peer reviews, which could add to or subtract from the credibility of the PRA results;

  - The credibility of qualitative insights obtained from the study;

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Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants

Appendices A, B, and C

Final Report

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

C.6 Mechanisms for PWR Reactor Vessel Depressurization Prior to Vessel Breach

The previous section addressed the range of thermodynamic loads to a PWR containment accompanying penetration of the reactor pressure vessel lower head by molten core debris and subsequent ejection of material into the containment atmosphere. These loads can present a significant challenge to containment integrity if penetration of the reactor vessel occurs at sufficiently high vessel pressure. For the three PWRs examined in this study, however, a substantial fraction of the severe accident progressions that started with the reactor vessel at high pressure depressurized before vessel breach. That is, many of the accident scenarios important to risk result in--by one means or another—a breach in the reactor coolant system (RCS) pressure boundary of sufficient size to reduce reactor vessel pressure below approximately 200 psi before reactor vessel lower head failure. An outcome of this result is that the uncertainties in high-pressure melt ejection loads are observed to have a relatively small impact on the overall uncertainties in reactor risk. This observation is a substantial change in results from those of preliminary analyses published in draft form in February 1987.

Unlike the BWRs examined in this study, the PWRs do not have a system specifically designed to manually depressurize the reactor vessel. Feed-and-bleed operations can effect limited depressurization if the necessary systems are operable. Many of the accident sequences leading to core damage in the three PWRs examined in this study, however, include combinations of failures that render feed-and-bleed operations unavailable. This section addresses the other means by which the reactor vessel pressure may be reduced to levels below which high-pressure melt ejection loads do not threaten containment integrity:

- Temperature-induced failure of steam generator tubes,
- Temperature-induced failure of primary coolant hot leg piping or the pressurizer surge line,
- Failure of reactor coolant pump seals,
- Stuck-open power-operated relief valves (PORVs), and
- Manual (operator) actions to depressurize the RCS.

The estimated frequency of each of these events and their influence on reactor vessel pressure was incorporated in the accident progression analysis for the Surry, Sequoyah, and Zion plants. Manual depressurization was found to be ineffective for most PWR accident sequences because of limitations in the appropriate emergency procedures and the need for ac power to operate relief valves. This mechanism is, therefore, not discussed further. The manner in which the other hypothetical events were considered, the means of quantifying their likelihood, and illustrations of the impact they have on the results are discussed in the following sections.

C.6.1 Issue Definition

The general issue is the frequency with which PWR severe accident progressions involve a breach in the RCS pressure boundary of sufficient size to reduce the reactor vessel pressure below approximately 200 psia. The mechanisms for depressurizing the reactor vessel that are considered in the present analysis are those listed in the introduction above. The first two mechanisms involve temperature-induced (i.e., creep rupture) failures of RCS piping. In both cases, the heat source for such failures is hot gases transported from the core via natural circulation or exiting the RCS through the PORV. The natural circulation pattern may involve an entire RCS coolant loop if water in the loop seals has cleared. If the loop seals have not cleared, a countercurrent natural circulation flow pattern may be established within the hot leg piping, transporting superheated gases and radionuclides from the core region of the reactor vessel to the steam generators. Effective cooling of the steam generator tubes is not available in many of the accident sequences considered in this analysis because of depletion of secondary coolant inventory earlier in the accident. Decay heat from radionuclides deposited in the steam generator inlet plenum and inside the tubes may also contribute to local tube heating. In either case, natural circulation flow (if established) may be interrupted by the frequent cycling of the pressurizer PORV or by the accumulation (and stratification) of hydrogen in the reactor vessel upper plenum and hot legs. The specific parameter to be quantified is the frequency with which creep rupture of hot leg piping or steam generator tubes results from the transfer of heat from the core (via gas circulation) to RCS structures. The temperature-induced failures of interest here are limited to those that occur before reactor vessel failure.

Degradation and failure of reactor coolant pump seals may also result from overheating. In this case, overheating results from the loss of seal cooling water flow or loss of heat removal from the seal cooling water system. A number of potential "seal states" have been identified in reactor coolant pump performance studies, which result in a range of plausible leak rates from the reactor coolant system. The parameters to be quantified are the frequency of pump seal LOCAs, the relative likelihood of various leak rates that result from these failures, and the resulting value of reactor vessel pressure at the time of vessel breach.

The fourth mechanism considered in this analysis, stuck-open PORV(s), may result following the repeated cycling (opening and reseating) of the PORVs during the course of an accident. Such events have been observed (with relatively low frequency) during transient events in which plant conditions never exceed design basis conditions. PORVs have also been tested for their reliability to close after repeated cycles at design basis conditions. This issue considers the effect of beyond design basis conditions on the frequency with which PORVs fail to close after several cycles.

C.6.2 Technical Bases for Issue Quantification

Two of the four mechanisms, temperature-induced hot leg failure and steam generator tube ruptures, were presented to a panel of experienced severe accident analysts. Each panelist was asked to provide a probability distribution representing his estimate of the frequency of each event. Their judgments were to be based on current information, made available to each of the panelists, and their own professional experience. The panelists participating were:

Vernon Denny—Science Applications International Corp.,
Robert Lutz—Westinghouse Electric Corp., and
Robert Wright—U.S. Nuclear Regulatory Commission.

The individual distributions prepared by these panelists were then combined (i.e., an aggregate distribution was generated by averaging those of the three panelists) to develop a single distribution for application in the PRA. The methods used to aggregate individual panelists' distributions are described in Reference C.6.1.

The frequency of reactor coolant pump seal failures was addressed by an expert panel in support of the systems analysis for the PWRs (Ref. C.6.2). This panel's judgments were adopted for use in the accident progression event tree. Very limited data are available to support an assessment of the frequency of PORVs sticking open when subjected to severe accident conditions. A broad distribution was, therefore, assigned to the frequency of stuck-open PORVs. A summary of the technical bases for quantifying the frequency of RCS depressurization for each of these four mechanisms is given below.

Frequency of Hot Leg Failure

A case structure was established to consider a spectrum of plausible severe accident conditions for which the frequency of hot leg failures needed to be quantified. The case structure was formulated around accident sequences that represent a significant contribution to the total core damage frequency. The cases considered were:

Case 1: A classic TMLB'* scenario (station blackout). RCS pressure is maintained near 2500 psia by the continuous cycling of the PORV. The secondary side of the steam generator is at the steam relief valve setpoint pressure (approx. 1000 psia) and is depleted of coolant inventory. Reactor coolant pump seal cooling is maintained at the nominal flow rate.

Case 2: Station blackout sequence during which reactor pump coolant seals fail, yielding a leak rate equivalent to a 0.5-inch-diameter break in each coolant loop. The steam generator secondary coolant inventory is depleted and the auxiliary feedwater system is unavailable.

Case 3: Same as Case 2 except the steam generators maintain an effective RCS heat sink with auxiliary feedwater operating.

*Reactor Safety Study (WASH-1400) nomenclature for accident sequence delineation. The alphabetical characters represent compound failures of plant equipment leading to the loss of plant safety functions. The characters TMLB' represent a transient initiating event, loss of decay heat removal, and loss of all electrical power.

The technical bases used by the panelists for characterizing the frequency of temperature-induced hot leg failures for each case were dominated by calculations performed with various severe accident analysis computer codes and by several different organizations. Those cited by the panelists in their elicitations (Ref. C.6.1) included TRAC/MELPROG calculations of TMLB' scenarios in Surry (Ref. C.6.3), RELAP5/SCDAP calculations of similar accident scenarios (Ref. C.6.4), CORMLT/PSAAC calculations for Surry and Zion (Refs. C.6.5 and C.6.6), and MAAP calculations performed in support of the Ringhals Unit 3 PRA (Ref. C.6.7) and the Seabrook PSA (Refs. C.6.8 and C.6.9). Ringhals Unit 3 is a three-loop plant with an NSSS similar to that of Surry; Seabrook is a four-loop plant with an NSSS similar to those of Sequoyah and Zion.

Only two specific references were cited by the panelists regarding experimental data or other physical evidence of natural circulation and its effect on heating RCS structures. These were the natural circulation experiments sponsored by EPRI (Ref. C.6.10) and the results of post-accident examinations of the Three Mile Island Unit 2 core debris and RCS structures (Ref. C.6.11). Information from neither of these sources is believed to have significantly influenced the panelists' judgments on this issue.

The aggregate distribution for the frequency of temperature-induced hot leg failures are shown in Figure C.6.1 for Cases 1 and 2 outlined above. The probability that Case 3 would result in an induced hot leg failure was judged to be essentially zero. The distributions shown in Figure C.6.1 are displayed in the form of a cumulative distribution function (CDF); that is, the curve displays the probability that the frequency of an induced hot leg failure is not greater than a particular value. The likelihood of an induced hot leg failure, given a station blackout accident during which the reactor vessel pressure remains high (i.e., no reactor coolant pump seal LOCAs, stuck-open PORVs, etc.), is shown to be relatively high; the median frequency is greater than 95 percent. In contrast, lower reactor vessel pressures in Case 2 (with an early pump seal LOCA) make an induced hot leg failure unlikely; there is an 83 percent chance that a hot leg failure will not occur.

Frequency of Induced Steam Generator Tube Ruptures

Essentially the same information (results of several computer code calculations) were used to characterize induced steam generator tube rupture (SGTR) frequency. All three panelists agreed that the likelihood of an induced SGTR is quite low. The three panelists noted that temperature-induced tube ruptures are driven by the same phenomena that drive temperature-induced hot leg failure (natural circulation flow of hot gases from the reactor vessel); therefore, the frequency distributions are correlated. Two of the panelists believed that the frequency of SGTR is very small because of the assumption that the hot leg would fail first, and neither of their distributions for frequency of induced SGTR exceeded a value of 0.0005. The aggregate distribution (shown in Fig. C.6.2) is dominated by a single panelist, whose distribution was strongly influenced by consideration of pre-existing flaws in steam generator tubes, resulting in the assumption that SGTR might occur before hot leg failure.

Frequency of Induced Reactor Coolant Pump Seal LOCAs

The frequency of pump seal LOCAs of various sizes (corresponding to various pump seal states) was considered by a panel of experts as a systems analysis issue. Degradation mechanisms for reactor coolant pumps are highly plant- (or pump-) specific and can be quite complicated. Details of the analyses leading to the characterization of the various pump seal states and the corresponding spectrum of possible leak rates are not provided here but are available in the documentation of the expert panel elicitations (Ref. C.6.2). An indication of the potential importance of modeling pump seal LOCAs, however, can be found by examining the accident progressions for which the reactor vessel pressure remains at or near the system setpoint (e.g., station blackouts with no other breach in the RCS pressure boundary). In the Surry analysis, approximately 71 percent of these accident progressions result in a failure of the seals in at least one reactor coolant pump. Of these, roughly one-third are estimated to result in a large enough leak rate to depressurize the reactor vessel to less than approximately 200 psia prior to reactor vessel breach; another third result in leak rates small enough to preclude any significant depressurization. In the remaining one-third of the cases, the reactor vessel is at intermediate pressure (200–600 psia) at the time of vessel breach (Ref. C.6.12).

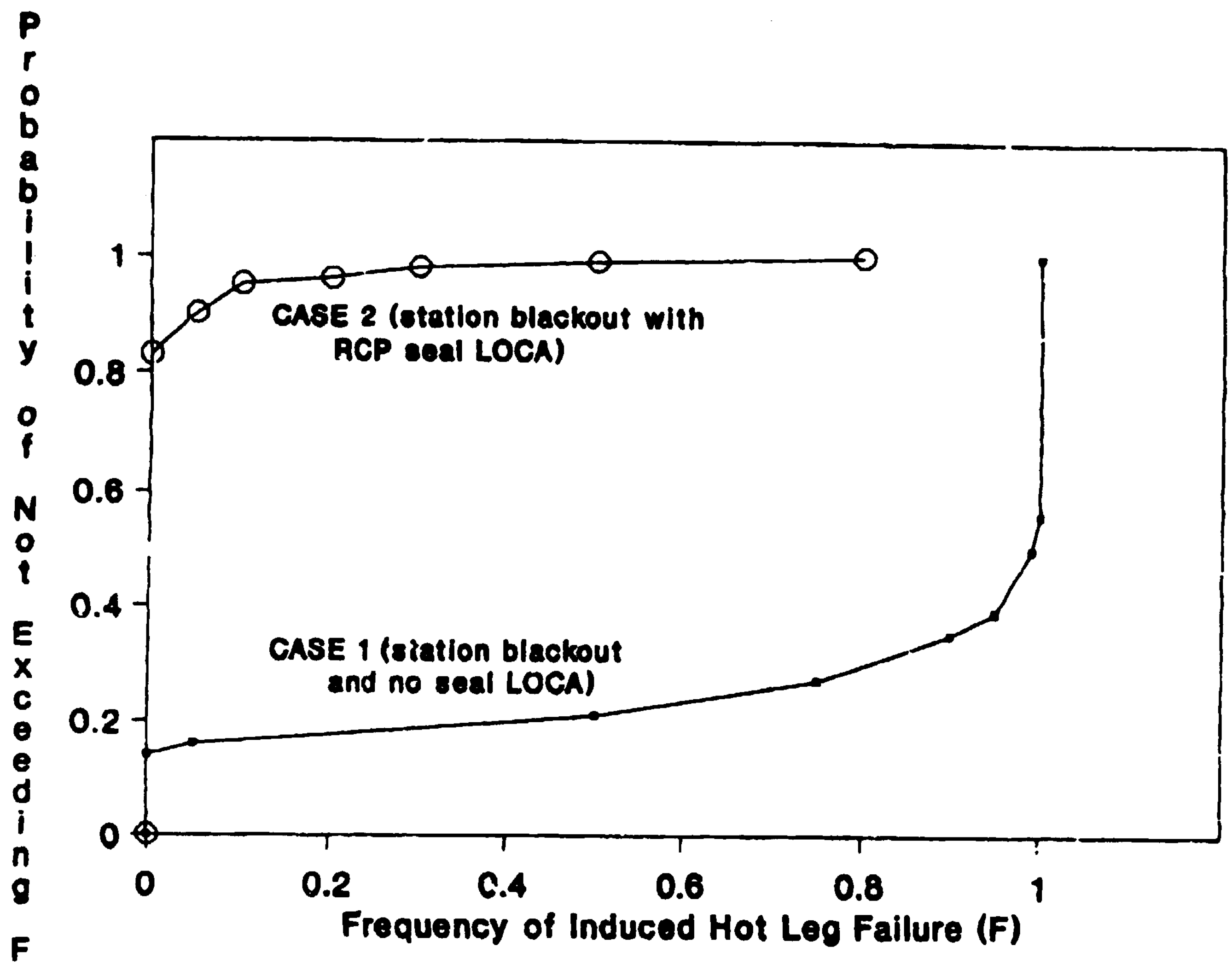

Figure C.6.1 Aggregate distribution for frequency of temperature-induced hot leg failure (Surry, Zion, and Sequoyah).

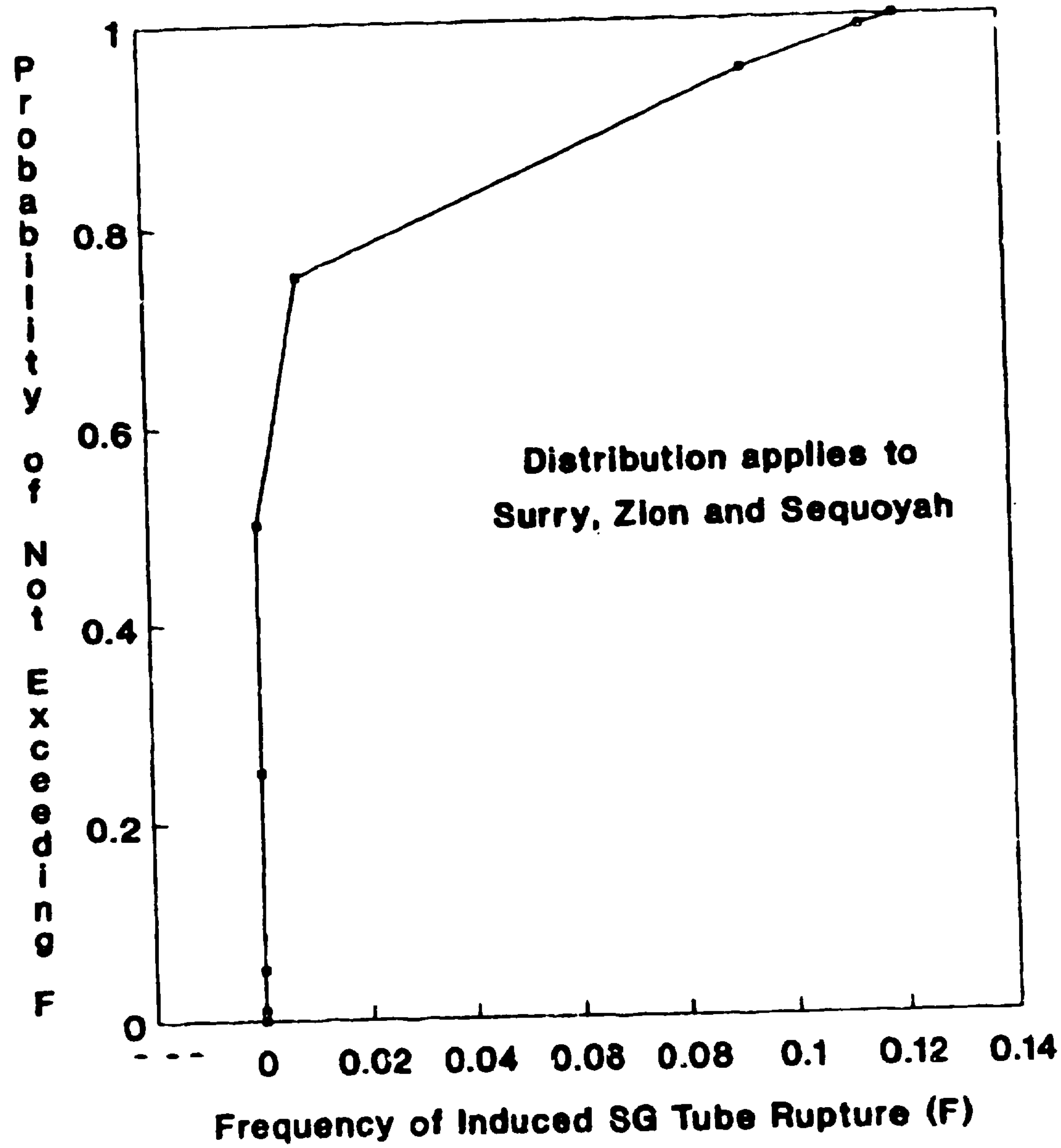

Figure C.6.2 Aggregate distributions for frequency of temperature-induced steam generator tube rupture.

Frequency of Stuck-Open PORVs

This issue was also addressed in the "front-end" analysis as an uncertainty issue (Ref. C.6.2). The RCS conditions under which PORVs will cycle after the onset of core damage, however, are expected to be significantly more severe than those for which the valves were designed and more severe than the conditions under which PORV performance has been tested. In lieu of specific analyses, test data, or operating experience, an estimate of frequency with which a PORV will stick open and an estimate for the resulting RCS pressure were generated as follows:

> The valve is expected to cycle between 10 to 50 times during core degradation and prior to vessel breach. Extrapolation of the distributions for the frequency of PORV failure-to-close from the front-end elicitations indicates an overall failure rate (for 10 to 50 demands) in the neighborhood of 0.1 to 1.0. A uniform distribution from zero to 1.0 was, therefore, used in the Surry and Sequoyah analyses.
>
> TRAC/MELPROG and Source Term Code Package (STCP) analyses were reviewed to characterize the rate at which a stuck-open PORV could depressurize the reactor vessel (Ref. C.6.12). The results of this review resulted in an estimate that there is an 80 percent probability that the reactor vessel pressure at the time of vessel breach will be less than 200 psia; in the remaining 20 percent of the cases, the vessel pressure will be at intermediate levels (200–600 psia).

C.6.3 Treatment in PRA and Results

The probability distributions for this issue were implemented in the PWR accident progression event trees. These trees (one for each plant) provide a structured approach for evaluating the various ways in which a severe accident can progress, including important aspects of RCS thermal-hydraulic response, core melt behavior, and containment loads and performance. The accident progression event tree for each plant is a key element in the assessment of uncertainties in risk; it considers the possibility that a particular accident sequence may proceed along any one of several alternative pathways (i.e., alternative combinations of events in the severe accident progression). The probability distributions for individual and combinations of events within the tree provide the rules that determine the relative likelihood of various modes of containment failure.

For the issue of reactor vessel depressurization, probability distributions for each of the mechanisms discussed above were incorporated in the accident progression event tree to determine reactor vessel pressure prior to vessel breach. As indicated in Section C.5, the containment loads accompanying vessel breach strongly depend on reactor vessel pressure. The load at vessel breach assigned to a particular accident progression, therefore, depends on the outcome of questions in the tree regarding reactor vessel depressurization. Selected results from the accident progression event tree analysis are summarized below.

The pressure history (as determined by the Surry accident progression event tree) for slow station blackout accident sequences* is summarized in Table C.6.1. This table shows the fraction of slow station blackout accident progressions for which the RCS pressure is at the PORV setpoint at high, intermediate, and low levels at the time the core uncovers and the time of reactor vessel breach.

A substantial fraction of the slow blackout accident progressions that start out with the RCS pressure at the PORV setpoint pressure are depressurized by one (or more) of the mechanisms described in Reference C.6.1 and result in a low pressure by the time of vessel breach.

A sensitivity study was performed to examine the effect of neglecting temperature-induced hot leg failure and steam generator tube ruptures on the observed results. Table C.6.2 summarizes the results of this study (presented in an identical format as Table C.6.1).

The results for pressure when the core uncovers are not affected by the change since temperature-induced hot leg failure and steam generator tube ruptures can only occur after the onset of core damage. The elimination of the possibility of these failures does affect the fraction of accident progressions involving reactor vessel breach at high pressure. The occurrence of high-pressure melt ejection is observed to roughly double in frequency.

*Slow station blackout accident sequences contribute more than one-half of the mean total core damage frequency for Surry. The results indicated for this group of accident sequences are not generally applicable to other Surry accident sequences or other plants.

Table C.6.1 Surry reactor vessel pressure at time of core uncovery and at vessel breach.

RCS Pressure (psia)	Fraction of Slow Blackout Accident Progressions With Pressure-P at the Time of: Core Uncovery	Reactor Vessel Breach
2500	0.54	0.06
1000—1400	0.13	0.10
200—600	0.33	0.19
<200	0.0	0.65

Table C.6.2 Surry reactor vessel pressure at time of core uncovery and at vessel breach (sensitivity study without induced hot leg failure and steam generator tube ruptures).

RCS Pressure (psia)	Fraction of Slow Blackout Accident Progressions With Pressure-P at the Time of: Core Uncovery	Reactor Vessel Breach
2500	0.54	0.25
1000—1400	0.13	0.10
200—600	0.33	0.19
<200	0.0	0.46

The increase in accident progressions resulting in vessel breach at high pressure is not observed to significantly affect the likelihood of early containment failure, however. Table C.6.3 shows the fraction of slow blackout accident progressions that results in various modes of containment failure (including no failure) for the Surry base case analysis and for the sensitivity analysis in which induced hot leg failures and steam generator tube ruptures were eliminated.

The insignificant change in results is largely attributable to the strength of the Surry containment and its ability to withstand loads as high as those estimated to accompany high-pressure melt ejection with a relatively high probability (refer to Section C.5).

Qualitatively similar results are observed for Sequoyah. Elimination of the potential for early reactor vessel depressurization by induced hot leg failure or steam generator tube rupture (via a sensitivity analysis) has a noticeable, but not dramatic, influence on the likelihood of high-pressure melt ejection. Table C.6.4 shows the fraction of Sequoyah accident progressions (for two important types of core melt accidents) that results in high-pressure melt ejection* for the base case analysis and the sensitivity analysis. In adjacent columns of this table are the fractions of the time that high-pressure melt ejection occurs and results in containment failure by overpressurization.

*The values shown only account for cases in which high-pressure melt ejection occurs in a cavity that is not deeply flooded. Cases in which the cavity is deeply flooded do not usually generate loads sufficiently large to threaten containment integrity.

Table C.6.3 Fraction of Surry slow blackout accident progressions that results in various modes of containment failure (mean values).

	Fraction of Slow Blackout Accident Progressions Resulting in Containment Failure Mode X	
Containment Failure Mode	Base Case Analysis	Sensitivity Analysis
Structural Rupture	0.01	0.01
Leak	0.01	0.01
Basemat Meltthrough	0.07	0.06
Containment Bypass	< 0.01	0.0
No Failure*	0.91	0.92

*Included in this category are accident progressions in which core damage is arrested in-vessel, thus preventing reactor vessel breach and containment failure. For Surry, these cases comprise approximately 60-65 percent of the "No Failure" scenarios.

Table C.6.4 Fraction of Sequoyah accident progressions that results in HPME and containment overpressure failure.

	Fraction Resulting in HPME Without a Flooded Cavity		Fraction of Columns (A) Cases in Which Containment Overpressure Failure Occurs	
Type of Core Damage Accident	(A) Base Case Analysis	(A) Sensitivity Analysis	Base Case Analysis	Sensitivity Analysis
LOCA	0.11	0.11	0.16	0.16
Station Blackout	0.16	0.21	0.20	0.21

As might be expected, no change is observed for the LOCA accident scenarios. Negligible changes are also observed for station blackout scenarios.

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Thompson Rept. Exh.: NRC, 1997b
(excerpts)

NUREG-1602

The Use of PRA in Risk-Informed Applications

Draft Report for Comment

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Division of Systems Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

3. INTERNAL EVENT LEVEL 2 PRA FOR FULL POWER OPERATIONS

This chapter provides attributes for performing a Level 2 probabilistic risk assessment (PRA) of a plant operating at full power. A Level 2 PRA evaluates containment response to severe accidents and determines the magnitude and timing of the radionuclide release from containment. Consequently, those PRA applications that deal with containment performance obviously need a Level 2 analysis as described in this chapter. A Level 2 analysis is also needed if the application requires that a numerical value for the frequency of a particular release be determined. Finally, if a particular PRA application requires estimates of offsite consequences and integrated risk, as, for example, in the calculation of the U.S. Nuclear Regulatory Commission (NRC) Safety Goal Quantitative Health Objectives (QHOs), then a Level 2 PRA coupled with a Level 3 PRA is needed. Accidents initiated by internal events including internal fires and floods are addressed in the following section. Accidents initiated by various external events are addressed in Chapter 5.

The primary objective of the Level 2 portion of a PRA is to characterize the potential for, and the magnitude and timing of, a release of radioactive material to the environment *given* the occurrence of an accident that results in sufficient damage to the core and causes the release of radioactive material from the fuel. To satisfy this objective, a quality Level 2 PRA is comprised of three major parts:

- A quality *Level 1 PRA*, which provides information regarding the accident sequences to be examined and their frequency. The attributes for performing the analyses associated with this aspect of a PRA are described in Chapter 2 and are not discussed further here.

- A structured and comprehensive *evaluation of containment performance* in response to the accident sequences identified from the Level 1 analysis.

- A quantitative *characterization of radiological release* to the environment that would result from accident sequences which breach the containment pressure boundary.

A detailed description of the attributes for conducting the technical analyses associated with each part is provided below.

The current state of knowledge regarding many aspects of severe accident progression and (albeit to a lesser extent) the state of knowledge regarding containment performance limits is imprecise. Therefore, an assessment of containment performance should be performed in a manner that explicitly considers uncertainties in the knowledge of severe accident behavior, the resulting challenges to containment integrity, and the capacity of the containment to withstand various challenges. The potential for a release to the environment is typically expressed in terms of the conditional probability of containment failure (or bypass) for the spectrum of accident sequences (determined from Level 1 PRA analysis) that proceed to core damage.

In addition to estimating the probability of a release to the environment, the Level 2 portion of a PRA should characterize the resulting radiological release to the environment in terms of the magnitude of the core inventory that is released, timing of the release, and other attributes important to an assessment of offsite accident consequences. This information provides (1) a quantitative scale with which the relative severity of various accident sequences can be ranked and (2) represents the 'source term' for a quantitative evaluation of offsite consequences (i.e., health effects, property damage, etc.) which are estimated in the Level 3 portion of a PRA.

One important aspect of a seismic event is that all parts of the plant are excited at the same time. This means that there may be significant correlation between component failures, and hence, the redundancy of safety systems could be compromised. The correlation could be introduced by common location, orientation, and/or vibration frequency. This type of "common cause" failure represents a unique risk to the plant that is reflected in a seismic PRA.

An additional consideration in the performance of a seismic PRA is the formation of both a well-organized walkdown team and a peer review team with combined experience in both system analysis and fragility evaluation. Ideally, the peer review should be conducted by individuals who are not associated with the initial evaluation to ensure ideally both technical quality control and technical quality assurance of the PRA results and documentation

Identification of Structures, Systems, and Components to be Included in the Seismic Analysis

The systems, structures and components (SSCs) modeled in the internal events PRA, internal fire PRA (Section 2.3), and internal flood PRA (Section 2.2) can be used in the identification of potential seismic induced initiating events, component failure modes, and accident scenarios. They provide the starting point for the identification of SSCs to be included in the seismic analysis. In addition, a review of the fire and flood analyses can help identify the potential for seismic-induced fires and floods. For example, failure of a heat exchanger or tank could lead to a flood that impacts other components. Similarly, rupture of an oil storage tank can cause a fire.

During the plant familiarization in preparation for performing a seismic PRA, plant documentation regarding equipment layout, design, and construction of the SSCs identified in the internal events PRA are typically reviewed. During this process, additional SSCs may be identified. During the plant walkdown, visual inspection of the equipment layout, component installation, and anchoring should identify SSCs whose failure could impact the risk of the plant. The plant walkdown is critical to identify as-designed, as-built, and as-operated seismic weak links in plants. Information is gathered to determine the significant failure modes of the SSCs and if the failure of an SSC would impact other equipment needed to mitigate the accident. For example, failure of a structure could cause failure of equipment nearby due to falling debris. More detailed attributes for a walkdown can be found in Sections 5 and 8 of the Electric Power Research Institute (EPRI) Seismic Margins Methodology (Ref. 5.4).

Initiating Events Analysis

Seismic-induced initiating events typically include transients, loss-of-offsite power (LOOP), and loss-of-coolant accidents (LOCAs). The postulated collapse of a major structure, such as the reactor building or turbine building, can be considered as an additional initiating event or as a basic cause for an initiating event that has been already identified in the internal events PRA. As mentioned previously, seismically induced fire and flood events can also be potentially identified. It is possible to have multiple initiating events for a given seismic event. This can be treated approximately by choosing the initiator with the worst impact from the standpoint of core damage probability and considering additional failures that are seismically induced. A systematic evaluation of the SSCs is performed to identify the causes of potential initiating events. In a manner similar to the way initiating events are grouped for an internal events PRA, the seismic failures can be grouped based on their impact on the plant. The results of the evaluation should produce a list of failures for each initiating event. The identified failures are then used to guide the quantification of the frequencies of the initiating events.

Hazard Analysis

In the 1980s, the methodologies for performing seismic hazard analysis were developed for the Eastern-U.S. sites by Lawrence Livermore National Laboratory (LLNL) (Ref. 5.5) and EPRI (Ref. 5.6). However, these seismic hazard curves by these two methodologies were significantly different for many of the eastern sites. As a result of the 1993 revision of the LLNL hazard curves (Ref. 5.7), either approach is currently considered to be acceptable. In 1993, an effort was also initiated to develop a method to produce more consistent seismic hazard curves (jointly supported by the NRC, EPRI, and the U.S. Department of Energy [DOE]). This recent development in seismic hazard analysis could also be used for future seismic PRAs. In the seismic hazard evaluation, site-specific soil conditions should be incorporated into the site-specific hazard curves to provide a true site-specific hazard evaluation. The potential for soil liquefaction should also be considered in a site-specific evaluation.

To quantify both the seismic hazard and component fragilities, a ground motion parameter needs to be selected. Traditionally, the peak ground acceleration or zero-period spectral acceleration has been used to represent the intensity of the earthquake hazard, which tends to introduce a significant uncertainty in the lower frequency range. To mitigate this problem, the average spectral acceleration is recommended for use since it expresses the ground motion intensity in terms of average response spectral values over the significant frequency range of interest for most structures and equipment (e.g., 5 Hz to 15 Hz). If an upper bound cutoff to ground motion at less than 1.5 g peak ground acceleration is assumed, sensitivity studies should be conducted to determine whether the use of this cutoff affects the delineation and ranking of seismic sequences.

Fragility Analysis

The fragility of a component or structure is defined as the conditional probability of failure given a value of the ground motion parameter. All the potential failure modes, both structural and functional, need to be examined to quantify the fragility value of a component. The sources of information that can be used in a fragility evaluation include the plant-specific design and test data, available experimental results, experience in past earthquakes (e.g., for offsite power loss), and generic fragility values from past studies.

Generic fragility parameters can be used in the initial screening of components and structures. However, the appropriateness of the generic fragility parameters has to be verified during the plant walkdown as well as by reviewing the documentation on component and structure fragilities. The high-confidence-and-low-probability (HCLPF) value can be used to screen components and structures without quantification of the seismic fault trees or event trees. Screening using a specified g-level for components and structure can be used to eliminate components with higher HCLPFs from further consideration in the PRA. However, if the core damage frequency (CDF) results indicate significant importance of components at the specified g-level, then components screened at this level should be added to the model and the results recalculated.

In the final PRA model, all components and structures that appear in the dominant accident cutsets should have site-specific fragility parameters that are derived based on plant-specific information, such as anchoring and installation of the component or structure. The methodologies for fragility analysis are discussed in a number of references, for example, NUREG/CR-2300 and EPRI NP-6041. It is desirable to incorporate the results of the latest available test data into the analysis and to also include aging effects in the component and structure fragility evaluation.

Seismic Model Development and Quantification

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February 15, 2000

MEMORANDUM TO: Those on the Attached List

FROM: Richard F. Dudley, Jr., Senior Project Manager /RA/
Decommissioning Section
Project Directorate IV & Decommissioning
Office of Nuclear Reactor Regulation

SUBJECT: TRANSMITTAL OF DRAFT FINAL TECHNICAL STUDY OF SPENT FUEL POOL ACCIDENT RISK AT DECOMMISSIONING PLANTS AND FEDERAL REGISTER NOTICE REQUESTING PUBLIC COMMENTS ON TECHNICAL STUDY

Attached is the "Draft Final Technical Study of Spent Fuel Pool Accident Risk at Plants." Also provided, for your information, is a copy of the *Federal Register* notice requesting public comments on the subject report.

Project 689

Attachments: As stated (2)

CONTACT: Richard Dudley, DLPM/NRR
301-415-1116

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Appendix 1 Thermal Hydraulics

1. Spent Fuel Heatup Analyses

Spent fuel heatup analyses model the decay power and configuration of the fuel to characterize the thermal hydraulic phenomena that will occur in the SFP and the building following a postulated loss of water accident. This appendix reviews the existing studies on spent fuel heatup and zirconium oxidation, the temperature criteria used in the analyses, and how it applies to decommissioned plants.

1.1 Spent Fuel Failure Criteria

Several different fuel failure criteria have been used in previously NRC-sponsored SFP accident studies. Benjamin, et. al used the onset of runaway fuel clad oxidation as the fuel failure criterion in NUREG/CR-0649 [Ref. 1]. This criterion was criticized because clad rupture can occur at a relatively low temperature causing a gap release. The consequences of gap release can be significant if the radioactive iodine has not yet decayed to insignificant amounts. SHARP calculations [Ref. 2] used the onset of clad swelling as an acceptance criterion for prevention of fuel failure. The onset of clad swelling leading to gap release occurs at approximately 565 °C, which corresponds to the temperature for 10-hour creep rupture time [Ref. 3]. A cladding temperature of 570 °C is used as a thermal limit under accident conditions for licensing of spent fuel dry storage casks.

The most severe fuel damage would be caused by rapid, runaway zirconium oxidation. This would lead to significant fission product release even after the gap activity has become insignificant. The onset of rapid oxidation may occur as low as 800 °C [Ref. 4]. Runaway oxidation can raise clad and fuel temperatures to approximately 2000 °C which corresponds to the melting temperature of zirconium. The release of fission products trapped in the fuel can occur at fuel temperatures of approximately 1400-1500 °C. Runaway oxidation starting in a high-powered channel could also propagate through radiative and convective heat transfer to lower power assemblies because of the large heat reaction in zirconium oxidation.

There are several other temperature thresholds that may be of concern in SFP accidents. The melting temperature of aluminum, which is a constituent in BORAL poison plates in some types of the spent fuel storage racks, is approximately 640 °C. No evidence was found that boron carbide would dissolve in the aluminum forming a eutectic mixture that liquefies at a temperature below the melting point of aluminum. However, if it is possible for a molten material to leak from the stainless steel spent fuel storage rack case, melting and relocation of the aluminum in the boron carbide-aluminum composite may cause flow blockages that increase hydraulic resistance. No realistic evaluation of melting and relocation of aluminum or aluminum/boron carbide eutectic has been performed.

Another concern is the structural integrity of the fuel racks at high temperatures. Several eutectic mixtures known from reactor severe accident research [Ref. 5] may be important in SFP accidents. As previously stated, the formation of an eutectic mixture allows liquification and loss of structural integrity for a mixture of materials at a lower temperature than the melting point of any of the component materials. Steel and zirconium form an eutectic mixture at approximately 935 °C. Steel

and boron carbide form a eutectic mixture at approximately 1150 °C. The steel racks may not be able to maintain structural integrity because of the sustained loads at high temperatures. Loss of rack integrity may affect the propagation of a zirconium fire.

If the gap radioactivity inventory is significant, then the spent fuel cladding temperature must be kept below 565 °C. If the consequences of aluminum/boron carbide relocation are acceptable, then 800 °C is a reasonable deterministic acceptance temperature, if uncertainties are less than the margin to 800 °C, and the effects of higher temperatures on the material are modeled. Otherwise, the temperature must be lower than the aluminum melting point (640 °C) or the aluminum/boron carbide eutectic melting point.

1.2 Evaluation of Existing Spent Fuel Heatup Analyses

In the 1980's, severe accidents in operating reactor SFPs were evaluated to assess the significance of the results of some laboratory studies on the possibility of self-sustaining zirconium oxidation and fire propagation between assemblies in an air-cooled environment, and also to assess the impact of the increase in the use of high density spent fuel storage racks on severe accidents in spent fuel pools. This issue was identified as Generic Safety Issue (GI) 82. Sandia National Laboratory (SNL) and Brookhaven National Laboratory (BNL) used the SFUEL and SFUEL1W computer codes to calculate spent fuel heatup in these studies. While decommissioned plants were not addressed in the study, many of the insights gained from these studies are applicable to decommissioned plants.

More recently, BNL developed a new computer code, SHARP, that was intended to provide a simplified analysis method to model plant-specific spent fuel configurations for spent fuel heatup calculations at decommissioned plants. Some of this work was built on the assumption used by SNL and BNL in their studies in support of GI 82.

1.2.1 SFUEL Series Based Analyses

Extensive work on the phenomena of zirconium oxidation in air for a SFP configuration was performed by SNL and BNL in support of GI 82. SNL investigated the heatup of spent fuel, the potential for self-sustaining zirconium oxidation, and the propagation to adjacent assemblies [Ref. 1, 6]. SNL used SFUEL and SFUEL1W computer codes to analyze the thermal-hydraulic phenomena, assuming complete drainage of the SFP water. In NUREG/CR-4982 [Ref. 4], BNL extended the SNL studies on the phenomenology of zirconium-air oxidation and its propagation in spent fuel assemblies. The SFUEL series of codes includes all modes of heat transfer, including radiation. However, radiation heat transfer may have been underestimated due to the assumed fuel bundle arrangement.

In NUREG/CR-0649, SNL concluded that decay heat and configuration are important parameters. SNL found that key configuration variables are the baseplate hole size, downcomer width, and the availability of open spaces for airflow. They also found that building ventilation is an important configuration variable.

The draft SNL report investigated the potential for oxidation propagation to adjacent assemblies. If decay heat is sufficient to raise the clad temperature in a fuel assembly to within approximately one hundred degrees of the point of runaway oxidation, then the radiative heat from an adjacent

assembly that reached the onset of rapid oxidation could raise the temperature of the first assembly to the runaway oxidation temperature. The report also discusses small-scale experiments involving clad temperatures greater than 1000 °C. SNL hypothesized that molten zirconium material would slump or relocate towards the bottom of the racks and consequently would not be involved in the oxidation reaction. NUREG/CR-4982 did not allow oxidation to occur at temperatures higher than 2100 °C to account for the zirconium melting and relocation. Otherwise, temperatures reached as high as 3500 °C. It was felt that not cutting off the oxidation overstated the propagation of a zirconium fire because of the fourth power temperature dependence of the radiation heat flux. The SFUEL series of codes did not model melting and relocation of materials.

In NUREG/CR-4982, BNL reviewed the SFUEL code and compared it to the SNL small-scale experiments and concluded that SFUEL was a valuable tool for assessing the likelihood of self-sustaining clad oxidation for a variety of spent fuel configurations in a drained pool. SNL reported the following critical decay times in NUREG/CR-0649 based on having no runaway oxidation. Critical decay time is defined as the length of time after shutdown when the most recently discharged fuel temperature will not exceed the chosen fuel failure criteria when cooled by air only.

700 daysPWR, 6 kW/MTU decay power per assembly, high density rack, 10.25" pitch, 5" orifice, 1-inch from storage wall

280 daysPWR, same as above except for 1 foot from storage wall

180 daysBWR, 14 kW/MTU decay power per assembly, cylindrical baskets, 8.5" pitch, 1.5" orifice

unknownBWR, high-density rack, SFUEL1W code was limited to computation of BWR low-density racks.

High-density racks with a 5-inch orifice are the most representative of current storage practices. A critical decay time for high-density BWR racks was not provided due to code limitations. Low-density and cylindrical storage rack configurations are no longer representative of spent fuel storage. All currently operating and recently shutdown plants have some high-density racks in the pool. For an assembly in a high-density PWR rack with a 5-inch orifice, a decay power below 6 kW/MTU did not result in runaway zirconium oxidation. All of these estimates were based on perfect ventilation (i.e., unlimited, ambient-temperature air) and burnup rates of 33 GWD/MTU. Currently, some PWRs are permitted to burn up to 62 GWD/MTU and some BWRs to 60 GWD/MTU. For fuel burnup of 60 GWD/MTU, the staff estimates the decay time for a bundle to reach 6 kW/MTU will increase from 2 years to approximately 3 years. Therefore, the staff expects the difference between critical decay times for PWRs and BWRs to decrease and that the BWR critical decay time for current burnups and rack designs would now be longer than the SNL estimate for high-density PWR racks. The SNL calculations also do not appear to have included grid spacer loss coefficients, which can have a significant effect since the resistance of the grid spacers is greater than the resistance of a 5-inch orifice. There is no mixing between the rising air leaving the fuel racks, and the relatively cooler air moving down into the pool. Including the grid spacer resistance, accounting for mixing and limiting the building ventilation flow to rated conditions, will result in the critical decay power to be less than 6 kW/MTU. The SNL calculations may have understated the effective radiation heat transfer heat sink due to the assumed fuel geometry in the calculations. A more realistic fuel configuration pattern in the SFP would give a

better estimate of the radiation heat sink and raise the critical decay power needed for significant oxidation.
While the studies in support of GSI 82 provided useful insights to air-cooled spent fuel assemblies, it is the opinion of the staff that they do not provide an adequate basis for exemptions. The studies were not meant to establish exemption criteria and lack sufficient information for all the parameters that could affect the decay time. Additionally, the reports are based on burnup values at that time. Since burnup values have increased, the results may not be directly applicable to today's spent fuel.

The general conclusions and the phenomena described in the studies assist in assessing issues for decommissioned plants. However, the calculated decay time values do not represent current plant operational and storage practices.

1.2.2 SHARP Based Analyses

In NUREG/CR-6451 [Ref. 7], BNL investigated spent fuel heatup that could lead to a zirconium fire at permanently shutdown plants. BNL developed a new computer code, SHARP (Spent Fuel Heatup Analytical Response Program), to calculate critical decay times to preclude zirconium oxidation for spent fuel. The code was intended to study thermal hydraulic characteristics and to calculate spent fuel heatup up to temperatures of approximately 600 °C. SHARP is limited to low temperatures since it lacks models for radiation heat transfer, zirconium oxidation, and materials melting and relocating. SHARP also lacks modeling for grid spacer losses and neglects mixing between the rising hot air and the falling cooler air in the SFP. BNL reported the following generic critical decay times using the SHARP code.

17 months for a PWR, high density rack, 60 GWD/MTU burnup; 10.4" pitch; 5" orifice
7 months for a BWR, high density rack, 40 GWD/MTU burnup; 6.25" pitch; 4" orifice

The above decay times are based on a maximum cladding temperature of 565 °C. The parameters listed with the critical decay times are generally representative of operating practices. Current fuel burnups in some plants, however, have increased to values higher than those used by BNL and perfect ventilation was assumed, which could lead to an underestimation of the critical decay times.

The SHARP code was not significantly benchmarked, validated or verified. The critical decay times above are shorter than those calculated in NUREG/CR-0649 and NUREG/CR-4982, particularly when the lower cladding temperature used for fuel failure and the higher decay heats used in the earlier analyses are taken into account. This appears to be driven in part, by the fact that the decay heat at a given burnup in the SHARP calculations is significantly lower than what is used in the SFUEL calculations. The staff has identified several areas that require code modifications, which will increase the calculated critical decay times. It is not adequate for use as technical bases by licensees without further code modifications and verification. NUREG/CR-6541 was intended as an assessment to steer rulemaking activities. The report was neither intended nor structured to provide a basis for exemptions. The staff does not rely on this study for heatup analysis information due to the code that the decay time conclusions were based upon.

1.3 Heatup Calculation Uncertainties and Sensitivities

The phenomenology needed to model spent fuel heatup is dependent on the chosen cladding temperature success criterion and the assumed accident scenario. Many assumptions and

modeling deficiencies exist in the current calculations. The staff reviewed the models to assess the impact of those modeling assumptions. Some of these uncertainties for the SFUEL series codes are further discussed in NUREG/CR-4982. For cases of flow mixing, decay heat, bundle flow resistance and other severe accident phenomena, additional information is provided here.

Calculations performed to date assume that the building, fuel, and rack geometry remain intact. This would not be a valid assumption if a seismic event or a cask drop damaged some of the fuel racks or the building. Rack integrity may not be a good assumption after the onset of significant zirconium oxidation due to fuel failure criteria issues discussed in Section 1.1. The building may also be hot enough to ignite other materials. Assuming that the racks remain intact is the most optimistic assumption that can be made about the rack geometry. Any damage to the racks or the building could significantly reduce the coolability of the fuel.

Previous SFUEL, SFUEL1W, and SHARP calculations, used in the resolution of GI 82 and decommissioning studies, used a perfect ventilation assumption. With the perfect ventilation assumption an unlimited amount of fresh, ambient-temperature air is available. This assumption would be valid if the building failed early in the event or if large portions of the walls and ceilings were open. If the building does not fail, the spent fuel building ventilation flow rate would dictate the airflow available. Mixing between the rising hot air and the descending cooler air in the spent fuel pool is not modeled in the codes.

The spent fuel building ventilation flow rate is important in determining the overall building energy balance. Airflow through the building is an important heat removal mechanism. Most of the air would recirculate in the building and the air drawn under the racks would be higher than ambient temperature and, therefore, less heat removal would occur. Airflow also provides a source of oxygen for zirconium oxidation. Sensitivity studies have shown that heatup rates increase with decreasing ventilation flow, but that very low ventilation rates limit the rate of oxidation. Other oxidation reactions (fires) that occur in the building will also deplete available oxygen in the building. Zirconium-Nitrogen reaction modeling is not included in the SFUEL code and may have an impact on zero and low ventilation cases. GSI 82 studies concluded that the perfect ventilation assumption was more conservative than no ventilation because the oxidation reaction became oxygen starved with no ventilation. These studies did not consider the failure modes of the building under high temperature scenarios. Intermediate ventilation rate results were not studied and give longer critical decay times than the perfect ventilation case.

A key fuel heat removal mechanism is buoyancy-driven natural circulation. The calculated airflow and peak temperatures are very sensitive to flow resistances in the storage racks, fuel bundles and downcomer. The downcomer flow resistance is determined by the spacing between the fuel racks and the wall of the SFP. The storage rack resistance is determined by the orifice size at the bottom entrance to the fuel bundle. Smaller inlet orifices have higher flow resistance. As shown by SFUEL and SHARP calculations, changes in the rack-wall spacing and the orifice size over the range of designs can shift critical decay times by more than a year. The fuel bundle flow resistance is determined by the rod spacing, the grid spacers, intermediate flow mixers and the upper and lower tie plates. SFUEL and SHARP calculations have neglected the losses from the grid spacers, intermediate flow mixers and the tie plates. These flow resistances will be higher than those from the rack inlet orifice in some cases. Therefore, inclusion of this additional flow resistance may significantly extend the critical decay time for some cases. NUREG/CR-4982 concluded that the largest source of uncertainty was due to the natural circulation flow rates.

The downcomer and bundle inlet air temperatures and mass flow rates are important in determining the peak cladding temperature. The extent of flow mixing will determine the air temperatures at the downcomer and bundle inlet. The SFUEL and SHARP calculations assume a well-mixed building air space. The downcomer inlet temperature is set equal to the building temperature. This assumption neglects the mixing that occurs between the hot air rising from the bundles and the cooler air descending down the SFP wall. Computational fluid dynamics calculations performed by the NRC using the FLUENT code and Pacific Northwest National Laboratory using the TEMPEST code indicates that the well-mixed building is not a good assumption. The mixing that occurs between the cool air flowing down into the pool and the hot air flowing up out of the fuel bundles can significantly increase peak cladding temperatures. Even using different turbulent mixing models can affect the peak temperatures by approximately 100 °C. The calculations indicate that fully 3-dimensional calculations may be needed to accurately predict the mixing because unrealistic flow topologies in 2-dimensional approximations may overstate the mixing. The calculations also indicate that the quasi-steady state assumptions for conditions above the fuel rack may not be appropriate. Time varying temperature fluctuations on the order of 100 °C have been observed in 3D calculations.

Radiation heat transfer is important in spent fuel pool heatup calculations. Radiation heat transfer can affect both the onset of a zirconium fire and the propagation of a fire. Both the SFP loading pattern and the geometry of the fuel racks can affect the radiation heat transfer between adjacent bundles. Simple gray body calculations show that at clad temperatures of 800 °C, a temperature difference of 100 °C between adjacent bundles would cause the radiation heat flux to exceed the critical decay power of 6 kW/MTU. Therefore, the temperature difference that could be maintained between adjacent bundles is highly constrained by the low decay heat levels. SFUEL calculations performed by SNL and BNL included radiation heat transfer, but the radiation heat transfer was underpredicted since the spent fuel placement is two-dimensional and the hottest elements are in the middle of the pool with cooler elements placed progressively toward the pool walls. Heat transfer between hotter and cooler assemblies has the potential to be significantly higher if the fuel bundles were intermixed in a realistic loading pattern.

At temperatures below 800 °C, the SFP heat source is dominated by the spent fuel decay heat. SNL and BNL found that, for high-density PWR racks, that 6 kW/MTU was the critical decay heat level for a zirconium fire to occur in configurations resembling current fuel storage practices. At the fuel burnups used in the calculations, this critical decay heat level was reached after two years. Decay heat calculations in NUREG/CR-5625 [Ref. 8] were performed to be the basis for calculating fuel assembly decay heat inputs for dry cask storage analyses. These decay heat calculations are consistent with the decay heat used in SFUEL calculations. Extrapolation of the decay heat calculations from NUREG/CR-5625 to current burnups indicate that approximately 3 years will be needed to reach a decay heat of 6 kW/MTU. The extrapolation has been confirmed to provide a reasonable decay heat approximation by performing ORIGEN calculations that extend to higher burnup. The critical decay heat may actually be as low as 3kW/MTU when in-bundle peaking effects, higher density rack configurations and rated build ventilation flows are taken into account.

Several licensees have proposed using the current Standard Review Plan (NUREG-0800) Branch Technical Position ASB 9-2 decay heat model for SFP heatup calculations. Using ASB 9-2 decay heat with a "k factor" of 0.1 produces non-conservative decay heat values in the range of 1 to 4 years after shutdown. ASB 9-2 explicitly states that it is good for times less than 10,000,000 seconds (~ 116 days). The basis of ASB 9-2 is the 1971 ANS draft decay heat standard. The standard gives "k factors" to use beyond 10,000,000 seconds. The staff has found that a "k factor

of 0.2" will produce conservative decay heat values compared to ORIGEN calculations for the range of 1 to 4 years after shutdown.

1.4 Zirconium Oxidation Temperature

At temperatures below the onset of self-sustaining oxidation, decay heat of the fuel dominates the heat source. When zirconium reaches temperatures where air oxidation is significant, the heat source is dominated by oxidation. The energy of the reaction is 262 kcal per mole of zirconium. In air, the oxidation rate and the energy of the reaction is higher than zirconium-steam oxidation. Much less data exists for zirconium-air oxidation than for zirconium-steam oxidation. A large amount of data exists for zirconium-steam oxidation because of the large amount of research performed under the ECCS research program [Ref. 9]. If all of the zirconium in a full 17x17 PWR fuel bundle fully oxidizes in air over the period of an hour, the average power from the oxidation is 0.3 MW. The critical decay heat as determined with SFUEL is approximately 2.7 kW for the bundle. The oxidation power source would amount to approximately 60 MW if the whole core was burning. A 20,000 cubic feet per minute (CFM) airflow rate is needed to support that reaction rate based on 100-percent oxygen utilization. The SFUEL oxidation rate was modeled using several parabolic rate equations based on available data. SFUEL had limited verification against SNL experiments that studied the potential of zirconium fire propagation. BNL determined that although they could not find a basis for rejecting the oxidation rate model used in SFUEL, uncertainties in oxidation of zirconium in air could change the critical decay heat by up to 25-percent. It was found that the onset of runaway zirconium oxidation could occur at temperatures as low as 800 °C. Different alloys of zirconium had oxidation rates that vary by as much as a factor of four. Apparently it was found that oxidation in air was worse than oxidation in pure oxygen. This suggests that the nitrogen concentration can have a significant impact on the oxidation rate. Since the relative concentration of oxygen and nitrogen varies as oxygen is consumed this causes additional uncertainty in the oxidation rate. The oxidation was cut off at 2100 °C in the BNL calculations in support of GI 82. This was done to simulate zirconium clad relocation when the melting point of zirconium was reached. If the oxidation was not cut off, temperatures could reach as high as 3500 °C. It was felt the propagation to adjacent bundles was overpredicted if no cutoff temperature is used due to the fourth power dependence of temperature on the radiation heat fluxes.

The combustion literature cited in the June 1999 draft report shows that there is a large range in the temperature for zirconium ignition in air. Evidence cited from the literature states that bulk zirconium cannot ignite at temperatures lower than 1300-1600 °C. It is known from the extensive emergency core cooling system (ECCS) and severe accident research programs that zirconium-steam runaway oxidation occurs at temperatures below 1300 °C. Since oxidation in air occurs more rapidly than oxidation in steam, temperatures in this range are not credible for the onset of runaway oxidation in air. Correlations listed [Ref. 10] give ignition temperatures for small zirconium samples in the range of runaway oxidation computed by the SFUEL series codes when the geometry factors calculated from zirconium cladding are input into the correlations. Only one reference [Ref. 11] appears to be applicable to zirconium oxidation in sustained heating of fuel rods. In the referenced test, sections of zirconium tubing were oxidized at temperatures of 700 °C, 800 °C and 900 °C for 1 hour. The average oxidation rate tripled for each 100 °C increase in

temperature. This is consistent with the change in oxidation rates predicted by the parabolic rate equations examined in NUREG/CR-4982. The zirconium combustion literature reviewed for ignition temperature did not discount or provide alternate oxidation rates that should be used in the SFUEL calculations.

As discussed earlier, current operating plants bum fuel to higher levels than used in the evaluations. The BNL and SNL studies in support of GI 82 represented operating practices of the 1980's with bumup level around 33 GWD/MTU. In NUREG/CR-6451, BNL used burnup values of 40 and 60 GWD/MTU for BWRs and PWRs, respectively. While these values are closer to current operating practices, they still underestimate peak burnup values. Additionally, the decay heat at the same burnup level used in the SHARP analyses is significantly lower than that used in the SFUEL analyses. Given that burnup is an important parameter for determining the critical decay time, this is a significant change. The increase in burnup level will increase the critical decay time needed to ensure that air-cooling is sufficient to maintain the zirconium cladding below the oxidation temperature.

The BNL and SNL studies in support of GI 82 represented storage practices of the 1980's when plants were starting to convert to high-density storage racks. The studies did not address high density BWR racks, and the high-density PWR racks in the reports were not as dense as the designs used by many plants today. The higher density racking currently used will decrease the airflow available for heat removal. Therefore, lower decay heat values are needed to ensure that air-cooling is sufficient to maintain the zirconium clad below the oxidation temperature.

1.5 Estimated Heatup Time of Uncovered Spent Fuel

The staff recognized that the decay time necessary to ensure that air cooling was adequate to remain below the temperature of self-sustaining zirconium oxidation was a conservative criteria for the reduction in emergency preparedness criteria. Using the fact that the decay heat of the fuel is reducing with time, credit could be given, if quantified, for the increasing length of time for the accident to progress after all water is lost from the SFP. The staff sought to quantify the decay time since final shutdown such that the heatup time of the fuel after uncovery was adequate for effective protective measures using local emergency response.

The heatup time of the fuel depends on the amount of decay heat in the fuel, and the amount of heat removal available for the fuel. The amount of decay heat is dependent on the bumup. The amount of heat removal is dependent on several variables, as discussed above, that are difficult to represent generically without making a number of assumptions that may be difficult to confirm on a plant and event specific basis.

For the calculations, the staff used a decay heat per assembly and divided it equally among the pins. It assumed a 9X9 assembly for the BWRs and a 17x17 assembly for the PWRs. Decay heats were computed using an extrapolation of the decay power tables in NUREG/CR-5625 [Ref. 8]. The decay heat in NUREG/CR-5625 is based on ORIGEN calculations. The tables for the decay heat extend to burnups of 50 GWD/MTU for PWRs and 45 GWD/MTU for BWRs. The staff recognizes that the decay heat is only valid for values up to the maximum values in the tables, but staff ORIGEN calculations of the decay power, with respect to burnup for values in the table, indicate that extrapolation provides a reasonable and slightly conservative estimate of the decay heat for burnup values beyond the limits of the tables. Current peak bundle average burnups are approximately 50 GWD/MTU for BWRs and 55 GWD/MTU for PWRs. The BWR decay heat was

calculated using a specific power of 26.2 MW/MTU. The PWR decay heat was calculated using a specific power of 37.5 MW/MTU. Both the PWR and BWR decay heats were calculated for a bumup of 60 GWD/MTU and include an uncertainty factor of 6 percent.

The staff has also considered a scenario with a rapid partial draindown to a level at or below the top of active fuel with a slow boiloff of water after the draindown. This could occur if a large breech occured in the liner at or below the top of active fuel. Section 5.1 of NUREG/CR-0649 analyzes the partial draindown problem. For the worst case draindown and a lower bound approximation for heat transfer to the water and the building the heatup time slightly less than the heatup time for the corresponding air cooled case. More accurate modeling could extend the heatup time to be comparable to or longer than the air cooled case.

Calculations, assuming an instant draindown of the pool and air-cooling, only show a heatup time to fission product release of 10 to 15 hours at 1 year after shutdown. The worst case partial draindown could release fission products in 5 to 10 hours at 1 year after shutdown.

1.6 Critical Decay Times to Reach Sufficient Air Cooling

Based on the above discussion, the staff concludes the following with respect to critical decay times. Calculations using the SFUEL code in support of GI-82 have determined a critical specific decay heat of 6 kW/MTU is needed for the onset of runaway zirconium oxidation. The 6 kW/MTU estimate calculated using SFUEL in a high-density storage rack configuration is reasonable and is based on the best calculations to date. However, this estimate is based on perfect ventilation conditions in the building and lower density rack configurations than exist today.

For high bumup PWR and BWR fuel, the staff estimates it will take approximately 3 years to reach the critical decay heat level cited in NUREG/CR-4982. Better modeling of flow mixing and accounting for the grid spacer and tie plate flow resistance could reduce the critical decay power level and increase the critical decay time beyond 3 years, but this may be counterbalanced by increased radiation heat transfer from realistic fuel bundle loading. Other assumptions, such as imperfect ventilation, could extend the critical decay time for the onset of a zirconium fire by 1 to 2 years. The critical decay heat may actually be as low as 3 kW/MTU when peak to average rod bundle peaking effects and higher density rack configurations are taken into account. Accounting for imperfect ventilation and higher density spent fuel storage in the racks, the staff estimates it will take approximately 4 to 5 years to reach a decay heat of 3 kW/MTU for current plant fuel bumups. Plant-specific calculations using fuel decay heat based on the actual plant operating history and spent fuel configurations could yield significantly shorter critical decay times. Calculations performed using checkerboard fuel loadings indicate that the critical decay time can be reduced by one year or more if the highest power fuel is interspersed with low powered fuel or empty rack spaces.

1.7 Fire Propagation

The staff has not performed a sufficient amount of research to fully understand and predict the propagation of zirconium fires in a spent fuel pool. Based on the limited amount of work performed to date, the propagation is probably limited to less than 2 full cores at a time of 1 year after shutdown. This estimate is based on lowering the GI 82 estimate of the 6KW/MTU fire threshold to 3KW/MTU to account for building ventilation effects. The actual propagation will probably be dependent on the actual fuel loading configuration in the spent fuel pool. A long term experimental

and analytical research program would be required to reliably predict the propagation of a zirconium fire in a spent fuel pool.

1.8 Guidelines for Spent Fuel Pool Heatup Analysis

Licensees must use an appropriate evaluation model for any site specific spent fuel pool heatup calculations. An evaluation model includes one or more computer programs and other information necessary for application of the calculation framework to a specific transient or accident, such as mathematical models used, assumptions included in the programs, a procedure for treating the program input and output information, specification of those portions of the analysis not included in the computer programs, values of parameters and other information necessary to specify the calculation procedure.

The code(s) should be validated and documentation of the modeling, verification, validation and use of the computer programs should be maintained to document the adequacy of the computer program. Finally, the code should be developed and maintained under a Quality Assurance program that meets the requirements of 10 CFR Part 50, Appendix B.

Depending on the margins available, sensitivity or uncertainty analysis should be performed (and documented) to confirm that the combined code and application uncertainty is less than the design margin for the safety parameter of interest in the calculation.

Spent fuel pool heatup analyses should consider decay heat removal from both the fuel racks and the building. An accurate determination of fuel cladding temperatures in the spent fuel pool requires fluid flow and heat transfer analyses. The primary components of a heatup anlaysis are described in the paragraphs that follow.

The spent fuel pool heat source is determined by the decay heat in the spent fuel. The analysis should use methods that are appropriate for the fuel burnup and decay time. The lowest possible decay heat input can only be achieved by accurately tracking the burnup history of individual spent fuel pool bundles. The method for calulating the spent fuel pool decay heat including its uncertainty should be justified.

The fluid conditions immediately above the spent fuel racks are determined by the heat removal from the spent fuel racks to the outside of the building. This is primarily determined by the building ventilation flow rate. Heat transfer through the walls can also be important at low ventilation rates. Heat removal from the top of the fuel racks to the bulk building atmosphere is primarily determined by buoyancy driven flows. Radiation heat transfer can also be significant. A steady state solution may not exist for the problem being analyzed. Time dependent variations must be considered in the analysis if time averaging is used in order to use a steady state approximation. Spatial variations must also be considered if spatial averaging is performed to simplify the analysis. The choice of a turbulence model must be justified and its impact on the overall calculation uncertainty must be evaluated.

Heat removal from the spent fuel pool racks is governed by the fluid conditions immediately above the fuel racks and buoyancy driven natural circulation in the racks. The heat removal rates are determined by the balance between buoyancy driving forces and the flow resistance of the downflow area and the fuel racks. Downflow in low powered spent fuel bundles should be considered and accounted for. This can be very important in densely packed spent fuel pools with

little downcomer area available for downflow. Calculations should use wall friction factors and additive loss coefficients (including those due to orifices and grid spacers) that are appropriate for both the flow regime and the geometry.

The staff's experience suggests conduction, convection and radiation heat transfer can all be important in spent fuel pool rack heatup calculations. Neglect of any heat transfer mode should be justified. Convective heat transfer coefficients should be appropriate for both the flow regime and the geometry.

Certain phenomena will occur as peak temperatures increase and should be considered for in the analysis. Experimental data has shown that clad ballooning will occur if cladding temperatures exceed temperatures of approximately 560 °C for longer than 10 hours. The temperature threshold will be lower for longer thermal loading times. If clad ballooning is expected additional flow losses may occur. Many spent fuel pool racks use BORAL plates for criticality control. Aluminum melts at approximately 640 °C. Heat transfer calculations within the rack should predict the temperature of any aluminum in the rack. If the temperature of any aluminum in the racks is predicted to exceed its melting temperature the consequences of the melting and relocation must be analyzed. Possible consequences of aluminum melting and relocation include flow blockages and criticality. Zirconium oxidation in air can have a significant effect on heatup calculations at temperatures above 600 °C. Zirconium oxidation must be modeled using and appropriate reaction kinetics model that is supported by experimental data.

The licensee must integrate all pieces of the analysis to determine if runaway zirconium oxidation will occur. The impact of uncertainties on the predicted temperatures must be evaluated and compared to the margin available in the calculation. The propagation of uncertainties through each part of the analysis must be properly treated.

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High burnup effects on fuel behaviour under accident conditions: the tests CABRI REP-Na

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Abstract

A large, performance based, knowledge and experience in the field of nuclear fuel behaviour is available for nominal operation conditions. The database is continuously completed and precursor assembly irradiations are performed for testing of new materials and innovative designs. This procedure produces data and arguments to extend licencing limits in the permanent research for economic competitiveness. A similar effort must be devoted to the establishment of a database for fuel behaviour under off-normal and accident conditions. In particular, special attention must be given to the so-called design-basis-accident (DBA) conditions. Safety criteria are formulated for these situations and must be respected without consideration of the occurrence probability and the risk associated to the accident situation. The introduction of MOX fuel into the cores of light water reactors and the steadily increasing goal burnup of the fuel call for research work, both experimental and analytical, in the field of fuel response to DBA conditions. In 1992, a significant programme step, CABRI REP-Na, has been launched by the French Nuclear Safety and Protection Institute (IPSN) in the field of the reactivity initiated accident (RIA). After performing the nine experiments of the initial test matrix it can be concluded that important new findings have been evidenced. High burnup clad corrosion and the associated degradation of the mechanical properties of the ZIRCALOY4 clad is one of the key phenomena of the fuel behaviour under accident conditions. Equally important is the evidence that transient, dynamic fission gas effects resulting from the close to adiabatic heating introduces a new explosive loading mechanism which may lead to clad rupture under RIA conditions, especially in the case of heterogeneous MOX fuel.

1. Introduction

The optimized use of nuclear fuel in pressurised water reactors (PWRs), and particularly the economic aspects of the reactor core management, entice the nuclear industry to change significant parameters of the nuclear reactor operating mode. Relying on very encouraging experience feedback concerning fuel behaviour under normal operating conditions, Electricité de France (EDF), the French electrical energy utility, has introduced: the increase of the UO_2 fuel discharge burnup (from 33 000 to 47 000 MWd/t by mean assembly), the load follow operation (power variations according to the electrical grid requirements), as well as a new fuel, the MOX (mixture of uranium and plutonium oxides).

However, a study of the fuel behaviour under design basis accident conditions was not conducted for the increased discharge burnups. This particularly relates to the reactivity initiated accident (RIA) for which the postulated initiator is the ejection of a control rod bundle. For this accident, the main safety criteria currently in effect and intended to prevent accidental fuel dispersion, limit the energy injected during the accidental transient condition to 230 cal/g for fresh fuel and 200 cal/g for irradiated fuel.

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The EDF plan to request a new authorisation for burnup increase from 47 000 to 52 000 MWd/t (megawattday per ton of fuel) has led the safety authority to ask EDF to perform research on the behaviour of PWR fuel at high burnup in order to reassess the criteria and to evaluate the impact of the new reactor core managements. The IPSN (French Nuclear Safety and Protection Institute) was interested in participating to this programme.

The IPSN Department for Safety Research (DRS) was entrusted with this research programme through co-operative IPSN/EDF action, considering its competence as well as its unique experimental facilities.

2. Purpose of the tests

The postulated initiator of the PWR design basis reactivity accident is the ejection of a control rod bundle under the effect of the system pressure following a control rod housing rupture. The reactor's hot standby (280°C, 155 bar) was defined as an aggravating situation for this accident. The ejection of the control rods would lead to a temporary supercriticality and to a transient increase of the nuclear power in a group of fuel assemblies in the vicinity of the ejected bundle.

The danger associated to the reactivity accident power excursion resides in the rupture of the fuel rod cladding, followed by fuel dispersion that could finally lead to a steam explosion, the scattering of radioactive material and/or the loss of part of the reactor's core cooling possibility.

The CABRI REP-Na programme intends to study the early phase of the physical phenomena and the key mechanisms of the RIA transient. It mainly concerns the changes of the fuel (fissile material and cladding) induced by irradiation up to high burnup. Abrupt fuel overheating produces a mechanical interaction (Pellet clad mechanical interaction, PCMI) which reaches its maximum level in the near adiabatic phase, before the cladding temperature increases by thermal conduction. In a second phase, the cladding rapidly overheats and approaches the conditions to reach the critical heat flux (departure from nucleate boiling, DNB).

Three complementary parts characterize the IPSN research RIA programme for high burnup fuel:

- Global experiments in the sodium test loop of the CABRI reactor,
- Development of the transient thermo-mechanical fuel behaviour code SCANAIR,
- Measurement of specific high burnup properties for use in SCANAIR.

The characteristics of the sodium coolant allow to study the early PCMI phase of the transient sequence of events, i.e., the PCMI loading phase. As already mentioned, the evaluation of the failure risk during this early phase represented the major objective at the time when the programme was launched. From the beginning it was clear that this approach would not solve all the aspects of the high burnup issue, in particular, the failure risk related to DNB and post failure phenomena in the pressurized water environment.

The development of the SCANAIR code aimed at both, preparation and interpretation of REP-Na experiments and transposition to the reactor case.

Finally, three major separate effect programs have been adopted in order to understand the integral test results from the CABRI REP test program:

PROMETRA: an out-of-pile test program to measure mechanical properties of high burnup cladding under transient temperature and loading conditions.

PATRICIA: the determination of the cladding to water heat transfer correlation during rapid power transients.

SILENE: quantification of the kinetics of fission gas behaviour in the fuel during rapid power transients.

The data from these separate effect test programs are used to improve the modelling of the physical phenomena in the SCANAIR code. SCANAIR will then be validated against the REP-Na integral test data before being used for evaluating rapid reactivity transients in power reactors.

3. Test matrix

At the beginning of the programme, the fuel rod burnup and the transient energy deposition were the only parameters of the test matrix. Soon, through experiment feedback, other important parameters were identified such as the amplitude and the fine structure of corrosion as well as the energy injection kinetics (width of the power-pulse). Finally, nine tests, six UO_2 tests and three MOX tests (Table 1) were programmed.

4. Fuel evolution under reactor operation

The power operation of the fuel inside the reactor leads to important cladding and fissile pellet modifications.

Firstly, *the cladding* is submitted to a creep induced plastic strain under the effect of the PWR primary system pressure, 155 bar, and is plated against the fuel. This process of fuel/cladding 'gap closure' is actually ended around the middle of the second cycle (~1.5 years, ~20 000 MWd/t).

Henceforth, the fuel is in direct contact with its cladding and any rapid fuel expansion, with kinetics

Table 1
CABRI REP-Na test matrix and main results

Test (carried out)	Tested rod	Pulse (ms)	Energy at pulse end (cal/g)	Corrosion (μ)	RIM (μ)	Results and remarks
Na-1 (11/93)	EDF Grav5c, span 5, 4.5% U5, 64 GWd/t	9.5	110 (at 0.4 s) (460 J/g)	80, important initial spalling	200	Brittle failure at $H = 30$ cal/g, Hmax = 115 cal/g; fuel dispersion: 6 g including particles other than RIM, sodium pressure peaks
Na-2 (6-94)	BR3, 6.85% U5, 33 GWd/t	9.5	211 (at 0.4 s) (882 J/g)	4	–	No rupture, $\Delta\phi/\phi$ (max): 3.5% average value, FGR/5.54%
Na-3 [a] (10/94)	EDF, 4.5% U5, 53 GWd/t	9.5	120 (at 0.4 s) (502 J/g)	40	100	No rupture, $\Delta\phi/\phi$ (max): 2% max, FGR/13.7%
Na-4 (7/95)	EDF Grav5c, span 5, 4.5% U5, 62 GWd/t	75	97 (at 1.2 s) (404 J/g)	80, no initial spalling	200	No rupture, transient spalling, $\Delta\phi/\phi$ (max): 0.4% average value, FGR/8.3%
Na-5 (5/95)	EDF Grav5c, span 2, 4.5% U5, 64 GWd/t	9.5	105 (at 0.4 s) (439 J/g)	20	200	No rupture, $\Delta\phi/\phi$ (max): 1% max, FGR/15.1%
Na-6 (3/96)	EDF MOX, 3c, span 5, 47 GWd/t	~35	165 (at 1.2 s) (690 J/g)	40	–	No rupture, $\Delta\phi/\phi$ (max): 3.2% max, FGR/21.6%
Na-7 (2/97)	EDF MOX, 4c, span 5, 55 GWd/t	~40	175 (at 1.2 s) (732 J/g)	50	–	Rupture at 120 cal/g, pressure peaks, examination currently carried out
Na-9 [a] (4/97)	EDF MOX, 2c, span 5, 28 GWj/t	~40	228 (at 1.2 s) (953 J/g)	<20	–	No rupture, examination currently carried out
Na-8 (7/97)	Grav 5c, span 5, 4.5% U5, 60 GWd/t	75	106 (at 1.2 s) (443 J/g)	130, cladding presenting spalling	200	Rupture at 83 cal/g (or lower [b]) gas blow-out, no fuel dispersion, examination currently carried out

[a] Improved cladding i.e. low tin.
[b] Pertinence of signals at 45 cal/g to be investigated by post-test examinations.

exceeding the creep velocity of the clad material produces a strong mechanical interaction.

During the whole irradiation cycle, a corrosion process in the reactor forms a layer of zirconium oxide (ZrO_2) on the cladding external surface and introduces into the metal an important amount of hydrogen, proportional to the zirconia thickness. At high burnup (>50 000 MWd/t), it is possible to reach or even pass, for Zircaloy4 cladding, a zirconia thickness of 100 μm and ~800 ppm of hydrogen (Fig. 1).

An aggravating aspect of corrosion is produced when the oxide layer 'spalls' locally. The absence of oxide then produces a cold point towards which the hydrogen migrates and an accumulation of hydrides is formed at the cladding's surface (*blister*). The presence of a blister can lead locally to the total loss of the cladding's ductility (Fig. 2).

At very high burnup (~60 000 MWd/t) a very high degree of spalling was observed on certain assemblies fitted with standard, unimproved cladding. The new cladding materials, now introduced in the EDF plants, should not spall at this level of burnup; however, the precise mechanism of this phenomenon is not yet understood.

The fuel pellets are subject to a deep transformation under irradiation: cracking, accumulation of fission products and swelling. Among the fission products, the gaseous elements (Xe and Kr), retained under the form of nanometric bubbles on intra-, or inter-granular sites in the fuel, play a predominant role during fuel rapid overheating. At 60 000 MWd/t their STP volume is equivalent to about 1.6 cm^3/g, 16 times the volume of the fuel. Increased under rapid overheating, this gaseous volume presents considerable swelling, fragmentation and dispersion potentials.

Beyond about 45 000 MWd/t a peripheral zone is created at the fuel surface through a neutronic effect. The characteristics of this zone are a high plutonium content generating a very high local burnup rate, a submicronic grain size as well as very important porosity (~20%). This width of the rim-zone is in the range of 200 μm. This structure formation is called 'rim effect' and represents a phenomenon characterizing highly irradiated fuel. Fundamental studies are currently being performed and aim at the clarification of the rim effect, in particular, the subdivision of the fuel grains into submicronic fragments.

The MOX fuel shows specific differences compared to the classical UO_2 fuel. The MOX fissile material is plutonium. During the preparation of the MOX following the MIMAS procedure, a mother blend of uranium/plutonium mixed oxide is added to natural or depleted uranium oxide. Pelletizing and sintering of this powder mixture create an heterogeneous final product, with mixed oxide $(UPu)O_2$ agglomerates or clusters imbedded in the matrix of natural UO_2. During reactor irradiation, the fission occurs in the clusters which reach very high burnup rates compared to the nominal mean

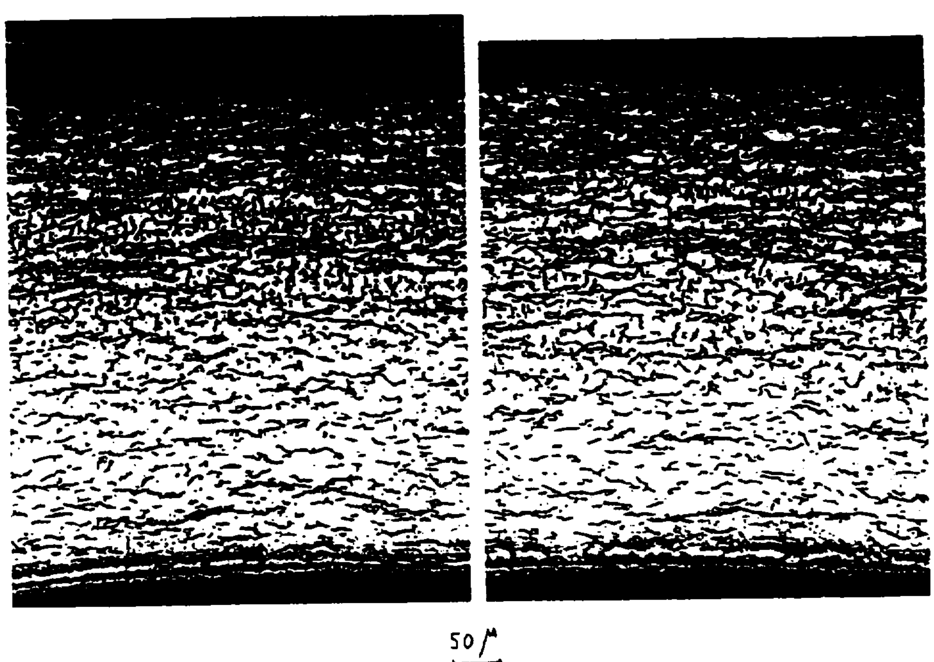

Fig. 1. Metallographic cut of the REP-Na4 rod cladding after CABRI test. The hydride plates are revealed by etching. The upper dark layer represents the ZrO_2 oxide layer with a thickness of about 80 μm (left). Large transient spalling occurred under this test (right).

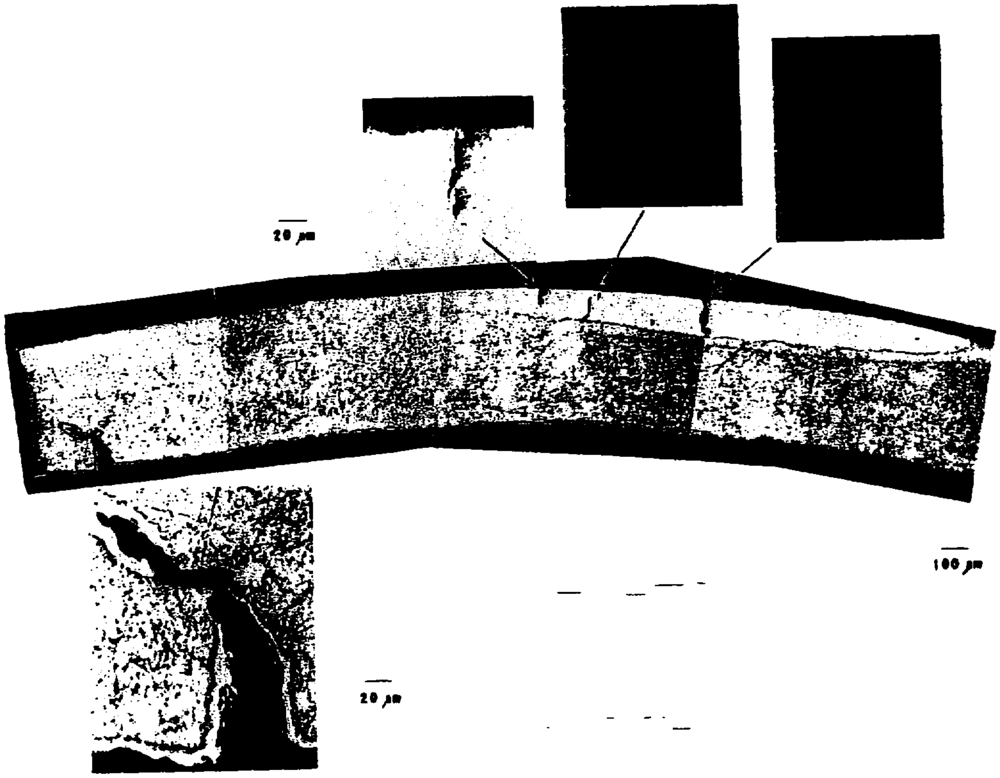

Fig. 2. The hydrogen migration towards a cold point of the cladding aggravates its embrittlement under irradiation in the reactor. The pre-existing cracking of the hydride phase can represent an incipient rod failure location under reactivity transient conditions. The photograph shows one of the REP-Na1 cladding blisters which are most probably the cause of the multiple ruptures during the test.

burnup (matrix average plus clusters). The structure and composition of the irradiated MOX clusters can be compared to those of the UO_2 fuel RIM, however, with a four to five times higher volume fraction.

5. Test results and phenomenological understanding

The main results currently available are presented in Table 1. The cladding rupture observed in the REP-Na1, Na7 and Na8 tests are remarkable and spectacular and contribute to the understanding of the failure mode and to the formulation of a failure criterion. The non-failure tests have produced valuable quantitative and qualitative results, for the understanding of physical mechanisms, and therefore for the development and validation of the SCANAIR code.

5.1. Mechanism and mode of rupture

In the first test of the matrix, the REP-Na1 test, a very early cladding rupture was recorded. This unexpected result was followed by a detailed metallographic examination programme and a series of calculations to identify the rupture conditions as well as its characteristics in order to conclude on the failure's cause and mechanism. The rupture aspect (Fig. 3) shows a purely brittle-fracture and the CABRI reactor measurements locate it at an instant which is described by the SCANAIR code calculation as a state where the RIM zone alone exceeds the nominal operating conditions. Details of the metallographic cuts show the presence of hydride accumulations (blisters) in the cladding. It is, therefore, possible to conclude that the rupture originated from a mechanical interaction due to the RIM effect, assisted by cladding embrittlement due to the presence of hydride (hydride assisted PCMI failure). It was demonstrated through the satisfactory rod behaviour during other UO_2 REP-Na matrix tests, that in case of moderate clad corrosion, the rod sustains PCMI charging even at a burnup greater than 60 000 MWd/t (REP-Na4, REP-Na5).

A second cladding rupture was observed in the REP-Na7 test, MOX test at 55 000 MWd/t. Examination of the tested rod is still to be carried out. However, a rupture mechanism such as during REP-Na1 appears unlikely, given the absence of spalling of the oxide layer. The sound cladding condition leads to the conclusion

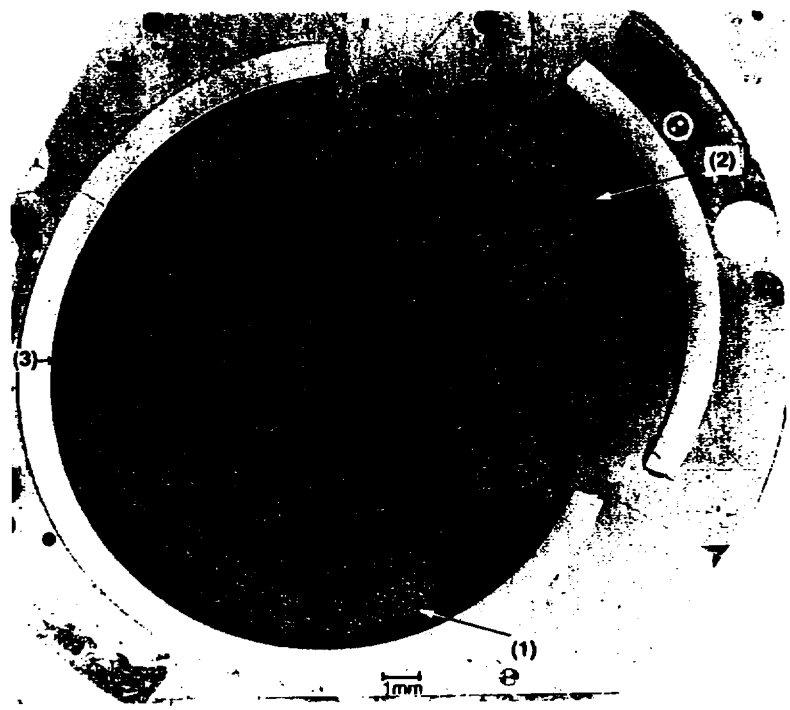

Fig. 3. Metallographic section (X4) of the REP-Na1 rod after test. The brittle aspect of the ruptures (perpendicular cracking) and the fuel fragmentation constitute outstanding facts of this observation. The numbers indicate locations of detailed examination: (1) RIM structure, (2) fragmented fuel, (3) intact cladding.

that the rupture mechanism is dominated by the contribution of fission gas to transient fuel swelling that could, in the case of MOX, be more important than for the UO_2 (also in discussion in Section 7). An examination programme of REP-Na 7 has been formulated with the aim to identify the rupture mechanism.

The cause and conditions of the rupture observed during the REP-Na8 test are currently the subject of investigations.

5.2. Cladding plastic strain

The fuel thermal expansion and the transient swelling are the two main factors contributing to cladding loading and cladding rupture occurs if the ultimate yield strength and the cladding's plastic strain capability are exceeded. In the CABRI tests without rupture, cladding strain is measured by profilometry. These examination results constitute valuable data for validation of the thermo-mechanical model of the SCANAIR code. Fig. 4 shows the REP-Na2 profilometry.

5.3. Fission gas driven fuel fragmentation

In all the CABRI REP-Na tests with significant plastic strain, a large fuel fragmentation zone is observed (Fig. 5). This fragmentation results from fuel grain decohesion under the effect of the fission gas fraction accumulated in micro bubbles in the intergranular zones. The bursting of the gas bubbles under the effect of fast transient heating leads to instantaneous increase of the fuel/clad contact pressure (PCMI) at high burnup when the fuel/clad gap is closed and also represents the driving force for grain separation. During the cooling process, when the cladding's permanent strain

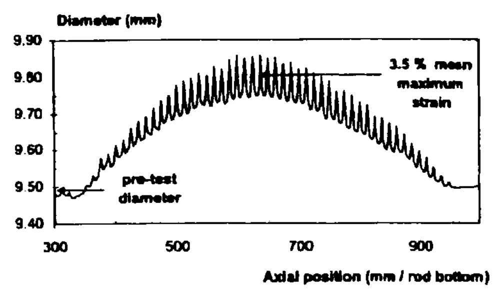

Fig. 4. REP-Na2 diametral straining over the length of the test rod. The shape of the curve traces the the axial power distribution in CABRI. The fine structure demonstrates each fuel pellets strain (hour glass type). This type of result provides precious elements for the SCANAIR code validation.

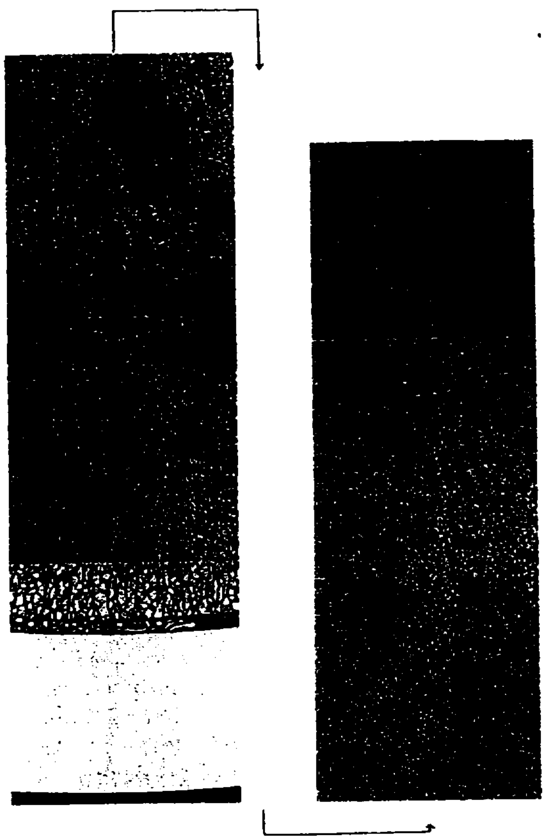

Fig. 5. Metallographic section of the REP-Na2 rod after CABRI testing. The grain decohesion and the loss of a large number of grains during preparation of the metallography demonstrate the fuel fragmentation. The fission gases accumulated in the fuel grain boundaries are the driving force of the transient fragmentation.

offers to the fuel an additional free volume, the grain-boundaries can open, producing the structure which is observed by ceramographic examination. This phenomenon of gas driven fuel expansion is a characteristic of the high burnup and can contribute, during cladding rupture, to the associated dispersion of fuel through the dynamics of the pressure relieve. The dispersion potential and its consequences are amplified in the RIM and in the MOX clusters due to the high local gas concentration and by the potential of emission of plutonium rich and submicronic particles.

5.4. Fission gas release

The fraction of fission gas, released during the test, is given in the results column of Table 1. Gases in intergranular sites alone are released in the very short RIA transient time. The activation of diffusion mechanisms releasing intragranular gases (majority fraction) only takes place for very high energy deposition (~200 cal/g). The amounts of released gas are significant, they increase the internal pressure of the rods not ruptured during the accident's first phase. The risk of creep-induced late rupture increases if the internal pressure exceeds the system's pressure. The release measurement results allow for validation of the gas behaviour models and to quantify the thermal and mechanical effects of the fission gas during the RIA transient [1].

5.5. Transient spalling

In several, highly corroded, REP-Na tests, a transient spalling of the oxide layer has been observed. In the very short RIA transient time, an important part of the oxide layer is detached from the cladding's metallic surface. In the case of an accident, this phenomenon introduces, even in the absence of cladding rupture, an important amount of debris into the reactor's coolant channels in a very short time and creates a risk of flow reduction and clogging. In addition, the cladding/water heat transfer could be reduced in the crucial phase of the accidental scenario when the fuel approaches critical thermal flux conditions and spalling oxide tiles influence the cooling conditions.

6. National and international co-operation

In this programme [2], the IPSN co-operates with several partners. EDF and FRAMATOME's active participation provides a stimulating complementarity [3]. The services and assessments issued from numerous laboratories of the CEA/DRN (neutronics, fuel codes, support tests, radiometallurgy) are essential to the programme's progress [4]. JAERI (Japan) is the senior international partner. In its NSRR test reactor, JAERI has been conducting RIA tests for many years and has also been observing high burnup cladding ruptures strongly associated to the cladding's corrosion level. Other observations (FGR, $\Delta\phi/\phi$) confirm and complete the REP-Na programme results [5]. NSI-KI (Kurchatov Institute, Russia) is a contractual partner and transmits, in the scope of the contract, its theoretical as well as experimental know-how (RIA programme in the IGR reactor in Kazakhstan) [6]. US-NRC has signed a co-operation agreement in June 1995 enabling it to access the CABRI REP-Na programme results as well as the support tests. Frequent and fruitful discussions, within the scope of this agreement, include American specialists from research and industry (ANL, INEL, BNL, PNL and EPRI) [7,8]. OECD-NEA has finally become the meeting ground for contacts with numerous other countries. The 'CSNI Specialist Meeting' in Cadarache, in September 1995 assembled more than 125 experts from 15 countries and this conference's proceedings [9]

provide a very complete view of the problematic of the light water reactor reactivity accident.

7. Discussion, conclusion and perspectives

Fig. 6 shows the RIA test database in terms of either maximum or failure enthalpy as a function of burnup of the test rods and underlines the contribution of the CABRI tests in the high burnup range.

This compilation, established by US-NRC, presents a large number of tests that should be sufficient to understand and validate the calculation codes. The following list indicates briefly the major non-prototypical conditions of the tests compared to RIA conditions in a PWR:

- CABRI: sodium coolant, low pressure: sodium cooling properties keep the clad temperatures low and low internal pressure mitigates the transient dynamic gas effect.
- NSRR: capsule tests, low pressure, low temperature, narrow pulse (~5 ms): the cladding remains during a significant time period below the brittle to ductile transition-temperature, the radial fuel temperature profile is anomalously peaking and critical heat-flux conditions are inadequately simulated.
- IGR: capsule test without instrumentation, low pressure, very wide pulse (>500 ms), low temperature, imprecise energy deposition: the radial fuel temperature profile is too flat.
- CDC/PBF: fuel not representative of the PWRs, most tests carried out in capsule, low temperature, low pressure: low fission gas retention and fuel restructuring (central hole formation) due to high power level during pre-irradiation.

The main drawback of this representation (Fig. 6) is the fact that it does not allow to fully assess the influence of clad corrosion and/or pulse width which are clearly identified as the high burnup key parameters. Nevertheless, examination of the data in Table 1 and Fig. 6 suggest that the fuel failure enthalpy is reduced significantly with fuel burnup. In addition, the REP-Na1 test underlines the unacceptable performance when oxide spalling and blisters are present in the cladding, i.e., a power excursion of low amplitude can result in fuel rod rupture. The original safety criteria of 230 (fresh fuel) and 200 (irradiated fuel) cal/g, presently being used, do not appear to be applicable to high burnup fuel.

It is suggested that the reduction in failure enthalpy with burnup, both for UO_2 and MOX, is due to the formation of very high burnup regions in each fuel type, i.e., the RIM structure in UO_2 and the clusters high in Pu (fissile material) in MOX. These very high burnup regions result in high concentrations of intergranular fission gas which produces fuel swelling during the transient and acts as an additional loading mechanism on the cladding. In the case of the MOX fuel, the high Pu clusters act similar to the RIM at high burnup with approximately five times the volume fraction of material

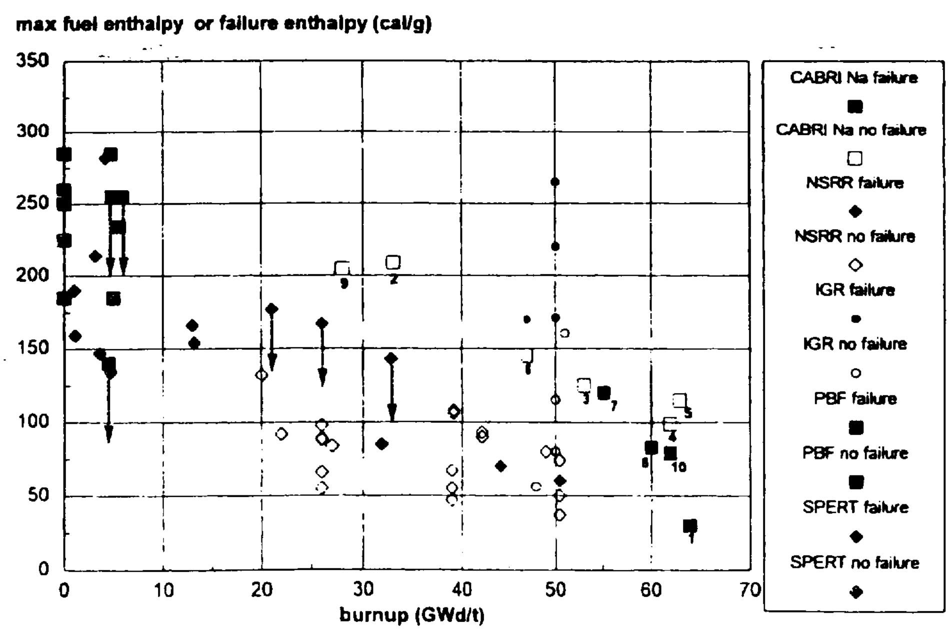

Fig. 6. A large number of experimental simulations of reactivity accidents has been carried out by several countries (US, Japan, Russia and France). The CABRI contribution includes all tests with burnup rates superior to 50 000 MWd/t as well as all of the irradiated MOX tests. The variety of results underlines the need to perform tests in realistic, representative conditions.

as the RIM in UO_2. This results in significantly higher loading for MOX rods than the UO_2 rods and increased failure potential at similar burnups and energy deposition levels as evidenced by the REP-Na7 test result, i.e., low failure threshold without hydride blisters.

It is further suggested that the increase in FGR and resulting high pressures with burnup observed in Table 1 may result in rod ballooning and rupture for rods that reach critical heat flux (CHF) during an RIA in PWR. This is not evident from RIA tests to date because they have all been tested in either sodium coolant or under low temperature and low pressure conditions where CHF is not easily achieved.

The study of this phenomenology requires PWR representative conditions. The sodium channel conditions, in the current CABRI facility, do not allow to reach this representativeness of the reactor situation. The diagram presented in Fig. 7 shows the cladding temperature evolution calculated by SCANAIR under sodium and pressurized water conditions for comparison. Only the phase 1 of the phenomenology could be studied by the REP-Na tests. The study of phase 2 requires experimental conditions which are representative of PWR conditions.

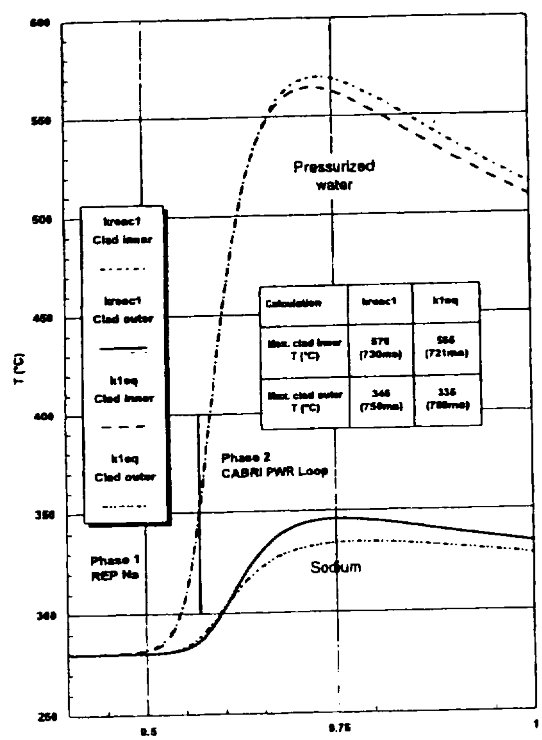

Fig. 7. The CABRI REP-Na tests were defined to study the PCMI phase of the accident phenomenology. Under sodium cooling conditions, the cladding temperatures remain comparatively low, as shown by the SCANAIR calculations. Under pressurized water cooling conditions, the departure from nucleate boiling leads to rapid clad overheating with risk of clad rupture by ballooning, as a consequence of the increase of the internal pressure by transient release of fission gas.

The installation into the CABRI facility of a pressurized water loop will enable the study of the whole spectrum of the accidental phenomenology (phases 1 and 2). The design and engineering work for this important transformation of the Cabri facility has been in progress for several years. The final decisions for this work should made in 1999 and the first experiment of a programme with international co-operation is expected to be performed at the end of year 2003. This programme will provide, for the future fuel design, the experimental database for the assessment and updating of the burnup dependent safety criteria for the design basis reactivity accident.

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