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

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

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

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EDITORIAL

The New Look of *Nuclear Safety*

As announced in our previous issue (Vol. 34, No. 2), this issue begins a semiannual publication frequency for the *Nuclear Safety* journal instead of the quarterly appearance of the recent past.

It must be apparent that the subscription cost per issue of the journal represents only a small fraction of the total costs of producing this publication. In fact, by government rules, the subscription price pays only for printing, binding, and mailing, so all the editorial costs must be borne by our sponsor. The reduction in publication frequency was mandated by a change in the sources of financial support of the journal, which went from joint funding by both the Department of Energy and the Nuclear Regulatory Commission (NRC) to funding by the latter alone. Although the level of support from the NRC has increased substantially, for which I am profoundly grateful to Dr. Eric Beckjord, Director of the NRC's Office of Nuclear Regulatory Research, it is nevertheless less than the total from the two combined sources in the past; this has led to the reduced publication frequency.

The change in appearance of *Nuclear Safety* as reflected in our new cover design is the outward manifestation of the more significant changes in the journal. The technical reviews and articles we publish form the backbone of the journal. It would be difficult to maintain the broad and thorough subject coverage that our title and history promise with only half as many articles per year. As regular readers of *Nuclear Safety* will remember, the journal used to carry, in addition to these technical articles, so-called current events items, in particular, the "Operating U.S. Power Reactors," "General Administrative Activities," and "Waste and Spent Fuel Management" as a regular part of each issue. These items constituted about 30 percent of the pages of each issue.

Given the new constraints on number of pages per year, I decided, in concert with the *Nuclear Safety* staff and with the support and concurrence of our NRC sponsors, to drop these columns from the journal. We did so for two reasons. The first is that, with publication only twice a year, the newsworthiness and interest of the current events material would, in any event, be greatly diminished. The second, and perhaps even more compelling reason for eliminating these columns, is that the space they occupied is freed up for printing more articles. Thus we will be able to carry not half, but rather 70 percent, as many pages of articles per year in two issues as had been carried in four same-size issues in the past.

We will, of course, make every effort to maintain the quality of these articles at as high a level as heretofore, and, in fact, the reduced number of articles per year will provide a motive for even more stringent acceptance criteria than we applied in the past. We will also significantly reduce the number of announcements of future technical meetings and courses, publishing only those which lie far enough in the future to be useful despite our infrequent publication schedule and which are of wide interest and importance.

As it has for thirty-five years, the journal will aim to serve its readers by presenting a wide spectrum of papers covering the entire range of disciplines that make up the field of nuclear safety. Quality papers will be solicited and accepted from around the world, with the only real criterion being that they be of interest and use to the U.S. and international nuclear safety communities and thus help establish and strengthen a safety culture that permeates every organization involved with nuclear matters so as to make nuclear energy and technology as safe and reliable as possible.

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

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

The Chernobyl Accident

Edited by E. G. Silver

Chernobyl Accident Management Actions

By A. R. Sich^a

Abstract: *Accident Management Actions taken during the first days after the Chernobyl accident either proved ineffective or were not fulfilled as reported by the Soviets at the International Atomic Energy Agency Meeting of Experts in Vienna in August 1986. Most significant to source-term analyses and estimates is that it is now believed that approximately 71% of the initial 190.3-tonne UO_2 fuel load was exposed to a high-temperature oxidizing environment because the core was neither covered with various materials thrown from helicopters to smother the fire nor was the core purged with (liquid) nitrogen. Both these actions were originally believed (on the basis of Soviet reports) to have effectively brought the crises to an end. These results seem to support earlier western far-field source-term estimates that significantly more volatile radionuclides may have been released as a result of the accident than reported by the Soviets in August 1986.*

Nuclear engineers have reached a consensus that the main cause of the Chernobyl accident (from the technical point of view) was reactor instability, primarily caused by design shortcomings of the RBMK-1000 reactor combined with violations of operating procedures (see endnote a, p. 18). More broadly, the accident was a direct result of fundamental design and administrative deficiencies. Strong management emphasis had been given to the importance of running a safety test before Unit 4 was shut down for scheduled maintenance. Management and operator actions placed a reactor of deficient design into progressively more unsafe configurations.

^aMassachusetts Institute of Technology.

CONTRADICTIONS AND CONCEALMENT

Various aspects of the accident that led to the destruction of Chernobyl Unit 4 have, to a greater or lesser extent, been studied and understood. The possibility of a large-scale imbalance in the neutron flux, the influence of thermo-hydraulic instabilities and pump cavitation, the faulty design of the emergency control (scram) rods, operator actions, etc., all played an important role in the development of the accident. Although certain unknown details may not change the general overview of the processes and main conclusions on the causes of the accident, their careful re-examination should be conducted with emphasis on the study of data that are now becoming available on the operation of Unit 4 just before the accident.

In contrast, the "Active" or "Source Term" Phase of the accident is investigated to a far lesser extent because little beyond scant descriptive information has been published or made available by the Russians or Ukrainians. The active phase of the accident is defined as the period from the initial destruction of the core, caused by the violent interaction of fragmented or molten fuel as it came into direct contact with the primary coolant and the subsequent steam explosion to the puzzling and very sharp drop in the release of radionuclides into the environment approximately 10 days following the accident. It is this stage that determined the distribution and state of the fuel after the explosion(s) and consequently the local conditions during the activity releases of the Active Phase. An examination of Chernobyl releases in terms of

physical and chemical processes taking place within the reactor fuel during the Active Phase would be useful if a relationship could be established between these processes and the observed releases of radioactivity. In turn, it may be possible to determine whether observations during the Chernobyl accident have any relevance or applicability to source-term predictions for severe accidents at other types of reactor installations in the West.¹ Finally, if a reliable model can be established, it may be possible to account for discrepancies between the far-field measurements of fallout from the accident and values presented by the Soviets at Vienna. This would prove invaluable for radiological experts to determine the extent of contamination shortly after the accident and by extension for comparisons with the currently observed rise in children's thyroid cancers in Ukraine and Belarus.^{2,3}

Indeed, one of the frustrating consequences of the accident is the gap between the conclusions of scientific research and the realities of the effects of the accident on the population at large and upon the surrounding environment.⁴ Although some scientists and engineers have concluded that in certain cases radiation fallout was too low to have significant effects, in many villages there has already been a discernible and even alarming rise in oncological sicknesses, even though the latent period for these cancers is somewhat greater than the time since the accident (see endnote *b*, p. 18). In addition, although the Soviets are to be commended for displaying a great deal of candor at the International Atomic Energy Agency (IAEA) Meeting in Vienna in August 1986, fission-product release information presented there must be reviewed to determine which data may be reliably used to estimate release rates during the Active Phase of the accident. Because of circumstances and the general unpreparedness of the Soviets for an accident of this magnitude, in part, brought on by their relatively lax attitude toward nuclear safety,⁵ much of the information, including information on the source term, is presented in a roughly summarized fashion. Moreover, many gaps in the information render some of it (except general descriptive accounts of what may have occurred a few days after the accident) virtually unusable.

The advent of what some would call "true" *glasnost* in the wake of the Chernobyl accident has heralded disclosures of cover-ups regarding the magnitude and extent of the accident⁶⁻⁹ (see endnote *c*, p. 18). Belarusian, Russian, and Ukrainian scientists have intimated in the press and to some of their colleagues in the West that there truly are a number of Chernobyl accident consequences and much data that were not discussed in the initial Soviet report at Vienna (see endnote *d*, p. 18),

despite the apparent openness of the Soviets in describing the accident (see endnote *e*, p. 18); for instance, it is now known that several pages detailing large quantities of radionuclides deposited 100 km and more northeast of Chernobyl were removed from the Vienna report following directives from the Soviet Central Committee.^{10,11} The Soviet press in 1989 began carrying more and more articles calling for an investigation into the "real" causes and circumstances surrounding the accident, alleging that the environmental impact was "tremendously downplayed."¹² These allegations were confirmed in mid-1989 (more than 3 years after the accident) when a secret decree of the USSR Ministry of Health (June 27, 1986) ordering silence about the Chernobyl accident and its effects upon the populace and cleanup workers was rescinded.^{13,14} Decisive action followed Ukraine's independence declaration of Aug. 24, 1991, and its December 1 independence referendum. Almost immediately following the referendum, a special committee of the Ukrainian Parliament presented to Ukrainian and Russian State Prosecutors evidence for a top-level conspiracy to conceal the extent and severity of the accident.

One of the most persistent myths concerning the extent of the accident—and perhaps the most relevant factor for a proper source-term analysis—is the claim by the Soviets at Vienna that the 5020 tonnes of sand, clay, dolomite, boron carbide, and lead dropped by helicopters on top of the reactor during the Active Phase of the accident and a liquid-nitrogen purging of the core together succeeded in cooling, smothering, and sealing the core almost completely from the environment and stopping the further release of radionuclides.¹⁵ Although INSAG-1 provides a generally positive appraisal of these Accident Management [Mitigation] Actions (AMAs),^{16,17} serious doubts as to the validity of Soviet release estimates arose quite soon after the accident.^{4,18,19} Inexplicable in particular (and questioned by western experts) is the curious shape of the total activity release curve (known as the "bathtub" curve and reproduced as Fig. 1) obtained from Soviet data.²⁰ Also rather puzzling is that the supposed filtration provided by the materials thrown onto the core and cooling of the core with nitrogen seem to have been ineffective in impeding the release of volatile radionuclides—especially if one considers the far-field estimates to be more representative of Chernobyl releases.

In fact, as this article will detail, in contrast to what was reported by the Soviets in August 1986, the material thrown onto the core in an attempt to smother the burning fire and the purging of the core with (ostensibly *liquid*) nitrogen were not successful. The implication is that the core "burned" virtually uncovered during the Active

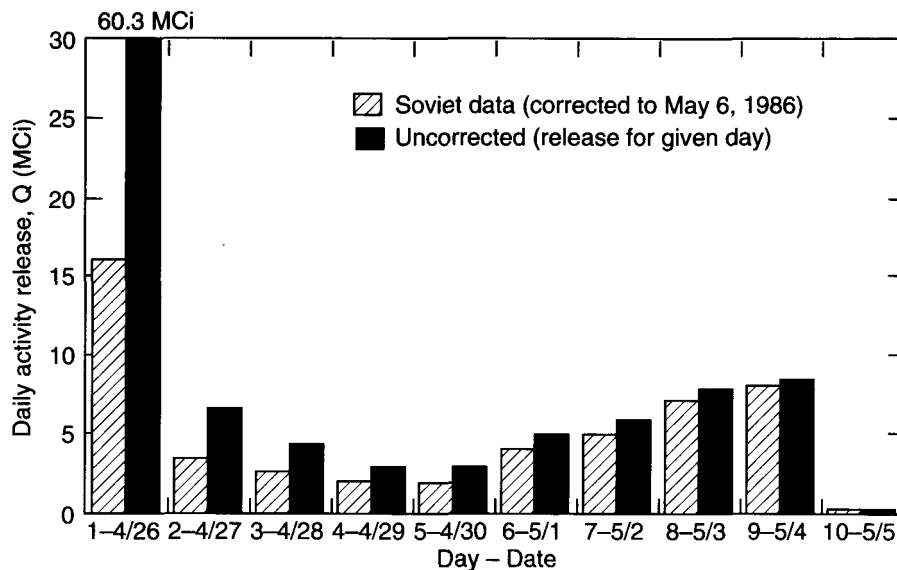


Fig. 1 Total Chernobyl Unit 4 releases over the Active Phase—the “bathtub” curve.²⁰

Phase of the accident—releasing radioisotopes into the upper levels of the destroyed reactor building and directly into the atmosphere. This not only calls into question Soviet and western analyses of Chernobyl source-term releases^{1,21} but also may have important ramifications for the planning of western accident management strategies. Moreover, the health impact to former residents of the 30-km exclusion zone surrounding the station (some of whom were not evacuated from this area until 8.5 days after the accident)²² must be re-examined in light of the greater releases suggested by this scenario.

CHRONOLOGY OF EVENTS AND ASSESSMENT OF MITIGATION EFFORTS TAKEN FOLLOWING THE INITIAL DESTRUCTION OF UNIT 4

This section will re-examine the effectiveness of Soviet AMAs to mitigate the consequences of the Chernobyl accident. Particular attention will be paid to the two actions that have a direct bearing on source-term releases, namely, the dropping of materials onto the core to smother the fire and the purging of the core with liquid nitrogen to cool it and hinder its ability to interact with surrounding structures. Of minor significance to source-term releases, but nevertheless interesting, are two other AMAs that will also be examined: (1) the release of

water from the pressure-suppression pool to mitigate the possible release of steam and enhanced transport of core materials if the molten core contacted the water, and (2) the construction of a flat-bed heat exchanger under the foundation of the Unit 4 reactor building to militate against a possible “China Syndrome.”

Appraisal of the Situation and Fire Containment

The tremendous task of mitigating the consequences of the Chernobyl accident began almost immediately after Unit 4 was destroyed (i.e., in the middle of the night of April 26, 1986). As a result of the explosion, significant quantities of reactor materials (in the form of large chunks of graphite, pressure tubes, and fuel assemblies) were hurled up to a distance of 150 m from the center of the reactor. Because of their initial high temperature and decay heat, these materials started more than 30 fires on the roofs of the Units 3 and 4 auxiliary buildings, the deaerator roof of Unit 3, and the machine hall above turbogenerator 7; unfortunately, much of the roofing of these buildings was pitched with tar (bitumen), which exacerbated the fires. Inside the damaged Unit 4 reactor building, other fires had started because fuel and lubrication oil pipes and containers were damaged. The immediate task, therefore, was to localize these fires to prevent them from spreading to the adjacent Unit 3 and eventually to douse them. By 02:15, fires on the roof of the

machine hall were localized; by 02:35, fires on the reactor and auxiliary buildings were put out; and by 06:35, all the fires in the vicinity had been extinguished.²²⁻²⁴

On Saturday morning (April 26th) the government formed a Government Commission charged with coordinating accident "liquidation" (see endnote *f*, p. 18) (mitigation) tasks along with the mobilization of resources necessary to carry out such tasks (see endnotes *g* and *h*, p. 18). Unfortunately, because initially no one seemed to know what had happened, nor was anyone sure of the extent of the damage (see endnote *i*, p. 18), incomplete information was being sent to Moscow as to the nature and extent of the accident (Ref. 11, p. 83). According to Serhij Vasylovich Shirokov (Head of the Division of Nuclear Power Stations in the Ukrainian Ministry of Energy and Electrification at the time of the accident, who was assigned as coordinator of mitigation efforts in Kiev), at first it was very unclear what had happened—mainly because Anatolyj Dyatlov, Deputy Chief Engineer of Units 3 and 4, and Viktor Brjukhanov, director of the station, refused to believe what had happened.²⁵ Although some plant personnel had risked their lives to climb to the roof and see what was going on (Ref. 11, pp. 81 and 82), they were either afraid to describe what they saw or were simply not listened to or believed. An attempt to cool the reactor with water seemed to fail (Ref. 11, p. 80), although through a greater part of the day on Saturday some station engineers still thought the core was being at least partially cooled by water. It was not until 10:00 p.m. on the day of the accident that a specialist from NIKIET (the RBMK design institute) flew over the reactor to assess the damage and reported that indeed it had been destroyed (see endnote *j*, p. 18). Apparently, first *official* recognition of what had occurred was around 3:00 in the afternoon.²⁶ It is more likely, however, that the extent of the damage was not known by the *members of the Government Commission* until some time in the evening after the reconnaissance mission was flown over the reactor (see endnote *k*, p. 18). In response, the Government Commission decided to "bomb" the reactor with various materials to smother the fires inside the central hall area (Ref. 11, pp. 106 and 107; Ref. 23, pp. 87 and 88).

Active Phase Accident Management Actions

Material Dumping to Smother the Reactor Fire. Most descriptions and appraisals of Soviet AMAs undertaken during the Active Phase to limit the release of radionuclides and to seal the reactor from the

environment do not fully reflect the reality of events during this period. One of the chief methods used to smother the fire was to dump thousands of tonnes of materials by helicopter into the Unit 4 reactor building. The materials dumped and reasons given by the Soviets for using these materials follow:

- | | |
|----------------------------|---|
| • Boron carbide, B_4C | To ensure against recriticality in the core. |
| • Dolomite, $MgCa(CO_3)_2$ | To generate carbon dioxide to provide a smothering gas blanket and contribute to the dissipation of core decay heat. |
| • A clay-sand mixture | To quench the fire and provide a filtration layer to limit radioisotope release. |
| • Lead ²⁷ | To absorb heat and provide a liquid layer that would solidify upon cooling to seal the core and shield the surrounding environment from ionizing radiation. |

Between April 27 and May 2 more than 1800 "bombing campaigns" dropped 5020 tonnes of friable materials into the reactor building—although some dumping of sand and clay continued through May 10 (Ref. 22, p. 83). [Helicopter logs indicate that, to the end of May 1986, 14 000 tonnes of solid materials, 140 tonnes of liquid polymers, and 2 500 tonnes of trisodium phosphate (a dust-suppressing liquid) were dropped into the damaged reactor building.]²⁸ Medvedev reports, "At 7 p.m. on May 1, [Boris Y.] Shcherbina [Deputy Chairman of the Soviet Council of Ministers and Chairman of the USSR Chernobyl Commission] announced that the volume to be dropped would be cut in half. There was reason to fear that the concrete structures supporting the reactor might not hold and that everything would collapse into the suppression pool, causing a thermal explosion and a massive release of radioactivity" (Ref. 24, p. 195) (see endnote *l*, p. 18). A summary of materials dropped is shown in Tables 1 and 2.

On the basis of these actions, many explanations have been provided to account for the unusual shape of the "bathtub" curve that were based partly on the data in

Table 1 Dates When Materials Were Dropped Into the Reactor Building of Unit 4^a

Date	Days after accident	Number of sorties flown	Mass of materials, tonnes	Mass-running sum, tonnes
April 26	0		—	—
April 27	1 ^b	44	150	150
April 28	2	93	300	450
April 29	3	186	750	1 200
April 30	4	?	1 500	2 700
May 1	5	?	1 900	4 600
May 2	6	?	420	5 020
May 3	7	0	—	—
May 4	8	0	—	—
May 5	9	0	—	—
May 6	10	0	—	—
By end of May	≈ 35	Total > 1 800	11 620	16 640

^aFrom helicopter pilot logs recorded in the research notes of Dr. Aleksandr Aleksandrovich Borovoi, Head of the Division of Radiation and Nuclear Safety of the "Shelter" Inter-Branch Scientific and Technical Center of the Ukrainian Academy of Sciences, Chernobyl, Ukraine.

^bNote that material deposition did not begin until at least 16:00 on April 27, or approximately 38 hours after the accident. (Private conversation with Serhii Vasyl'ovich Shirokov.)

Table 2 Materials Dropped Into the Reactor Building of Unit 4

Material	Chemical formula	Mass, tonnes
Boron carbide	B ₄ C	40 ^a
Dolomite	MgCa(CO ₃) ₂	800
Clay and sand	—	1 760
Lead	Pb	2 400 ^b
Other solid materials	—	9 000
Liquid polymers	—	140
Trinatriphosphatium	—	2 500
Total	—	16 640

^aThe boron carbide arrived Sunday night (April 27) (Shirokov).

^bThe lead arrived during the fifth day after the accident (April 30) (Shirokov).

Today, most agree that the materials thrown into the core in the days immediately following the initial explosions gave rise to the unusual shape of the "bathtub" curve (Ref. 23, pp. 90 and 91); for example, according to INSAG-1 (Ref. 16, p. 38), there are four possible explanations for the apparent increase in fission-product release rates beginning 5 to 6 days into the accident (May 1 to 2):

1. Once the initial wave of material deposition was stopped (on May 2), heat losses from the debris declined, the temperatures rose, and vaporization releases were enhanced.

2. Some increase in gas flow over the debris occurred that enhanced material removal by vaporization or enhanced chemical (oxidation) reactions.

3. The melting of deposited lead and the pyrolysis of dolomite came to an end, so heat losses from the debris dropped, the temperature of the debris rose, and vaporization release again increased.

4. Enhanced oxidation from some unidentified mechanism aided release.

Perhaps of greater interest than the increase in release rates on the fifth and sixth days is the sudden drop to almost negligible levels approximately 9 to 10 days after

Table 4.10 of the Soviet report presented at Vienna (Fig. 1), partly on ground and air radioactivity measurements, and partly on the assumption that the materials thrown into the reactor building covered the burning core.

the accident—signifying the end of the Active Phase. To date, no definitive explanation for this phenomenon has been available. One hypothesis, proposed also in INSAG-1, is

that the release accelerates because core debris reheats and liquefies. The required temperature for liquefaction is 2300–2900 K depending on the amount of unoxidized zirconium present in the debris. Vaporization accelerates upon debris liquefaction. The liquefied debris can relocate, eventually falling into the lower pipe runs where it can freeze. Continuing cooling flows of gas into the pipe runs prevent the quenched debris from either melting or significantly attacking concrete and steel structures in this part of the reactor (Ref. 16, p. 91) (see endnote *m*, p. 19).

Also put forward is the hypothesis that bombarding the core with materials may simply have done what it was intended to do (i.e., smother and seal the remains of the reactor)¹⁵ (see endnote *n*, p. 19).

Investigations conducted from 1986 to 1989 have shown that early ideas and descriptive models, concerning the extent to which damage may have occurred within Unit 4 after the accident, in most cases do not correspond to the actual conditions of the destroyed reactor. After a significant number of bore samples (about 70) had been taken of the core region, subreactor region, steam-distribution corridor, and a number of other areas, radiation field measurements were taken to determine which areas were safe enough to approach for closer inspection.^{29,30} Visual inspections of these areas were then conducted either by remote-control visual aids (video cameras, periscopes, or small robots), or, if approachable, directly by researchers armed with photo and video cameras. The most startling discovery was that the core region turned out to be practically empty (Fig. 2). Even more significant for the present discussion was the fact that, according to chemical analyses, the fuel-containing masses (FCMs or “lava” as it is commonly referred to) located in the lower regions of the reactor building contain only trace amounts of the materials thrown into the reactor in the attempt to smother it—rather, they have a material composition similar to that of a mixture of corium, the Lower Biological Shield (LBS) (see endnote *o*, p. 19), and other metal structures formerly located beneath the core.³¹ [The LBS, containing serpentine—a hydrous magnesium silicate $3\text{MgO} \cdot 2\text{SiO}_2 \cdot 2\text{H}_2\text{O}$ or $\text{Mg}_6(\text{Si}_4\text{O}_{10})(\text{OH})_8$, cast iron-pebble filler material, and stainless steel coolant piping surrounded by stainless steel plating—is the only structure in the vicinity that contains substantial quantities of magnesium, traces of which are found in the FCMs.]³²

In addition to chemical analyses, detailed examination of video clips of the damaged reactor taken from a

helicopter 2 to 3 days after the accident clearly shows that the red-glowing mass of (presumably) burning graphite (see endnote *p*, p. 19) is located away from the shaft of the reactor in the area just to the east of the southern spent-fuel pool on the level of the Central Hall (high-bay area). Little is visible of the reactor shaft itself because it is blocked by a significant amount of severely damaged “upper water and steam communication piping” (i.e., coolant piping) and the Upper Biological Shield (UBS) is positioned almost vertically on its side blocking most of the shaft. Finally, visual investigations conducted later showed that the vast majority of materials thrown into the reactor area formed a huge pile approximately 8 to 12 m high that partially covered the opening to the southern spent-fuel pool precisely where the “burning” mass was seen (see endnote *q*, p. 19) (see Figs. 2 to 4). Apart from the fact that accurate bombing of a small target from 250 to 300 m in a highly radioactive environment constituted an extremely difficult task for the pilots. Dispatch commands were radioed from an observer either on the roof of the Prypjat’s Communist Party Executive Committee building (*Miskvykonkom*) or from the hotel *Polissja* in the city (Ref. 11, pp. 107 and 123) (see endnote *r*, p. 19)—3 km away and facing into the sun for the first half of the day³³ (see endnote *s*, p. 19). Evidently, the helicopter crews, in an effort to smother the blaze, aimed for the burning mass—missing the reactor shaft completely. It was not until 1990–1991 that the Soviets reported the fact that few if any bags of materials had fallen on top of the core²⁸ (see endnote *t*, p. 19).^{34–36} It seems that the Soviets knew this quite early on (see endnotes *u* and *v*, p. 19) or at least could have deduced it from aerial photographs over the destroyed reactor. As supporting evidence for the renderings in Figs. 2 to 4, Fig. 5 is one of these photographs. Clearly visible is the UBS hanging inside the reactor shaft and the large pile of material thrown in the area of the southern spent-fuel pool. It is also clear from this photograph that no pile of materials (contained in large canvas and plastic bags) covers the UBS or the core shaft.

(Liquid) Nitrogen Purge of the Core Region. On the basis of a May 1 decision by the Governmental Chernobyl Commission, beginning early on May 6 (see endnote *w*, p. 19) the authorities started to “purge the core with nitrogen” by pumping it (using a station compressor) under the core through the “lower piping communication” (coolant channels) under the reactor to eliminate air ingress, to provide cooling for the core, and to prevent further oxidation (burning) (Ref. 15, pp. 39 and 40) (see endnotes *x* and *y*, p. 19). According to Medvedev,

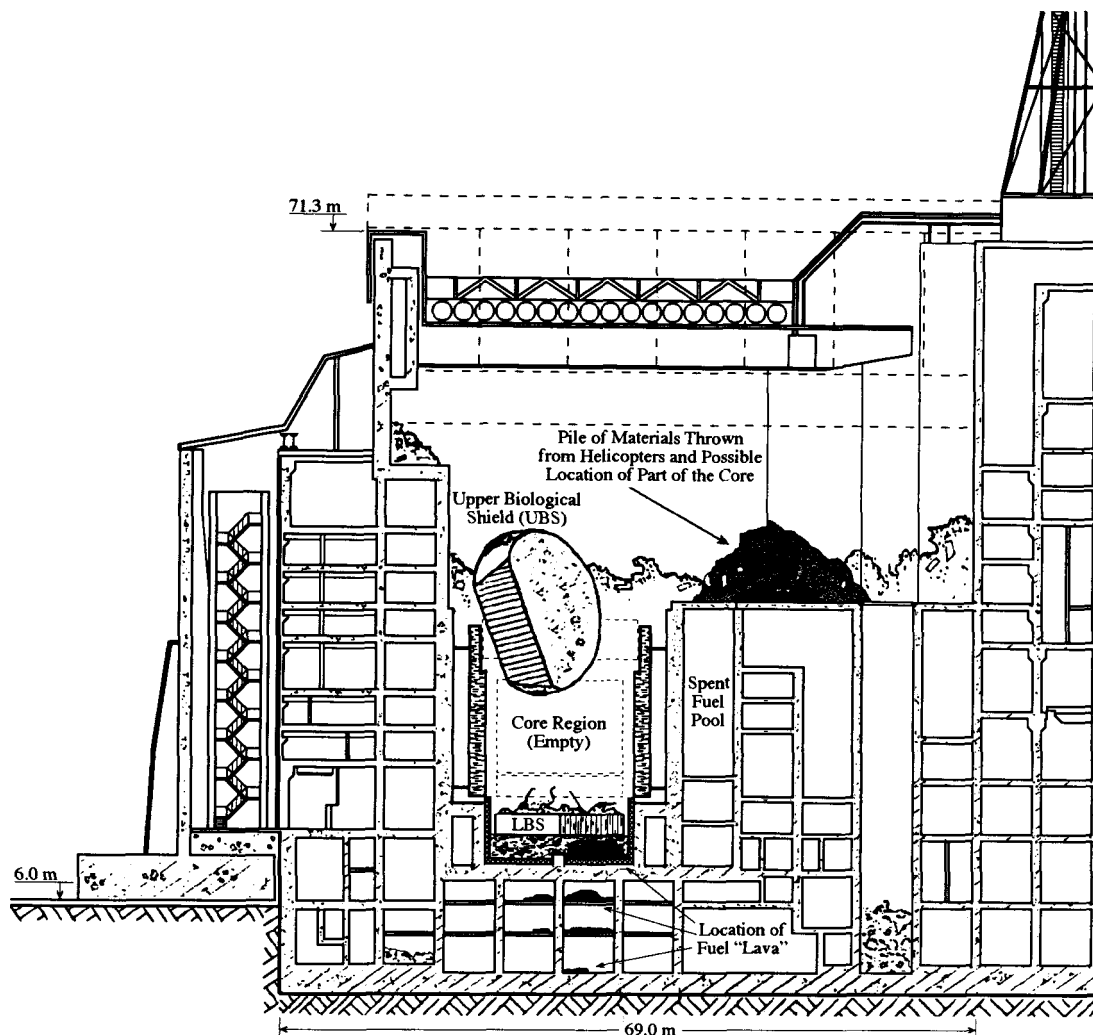
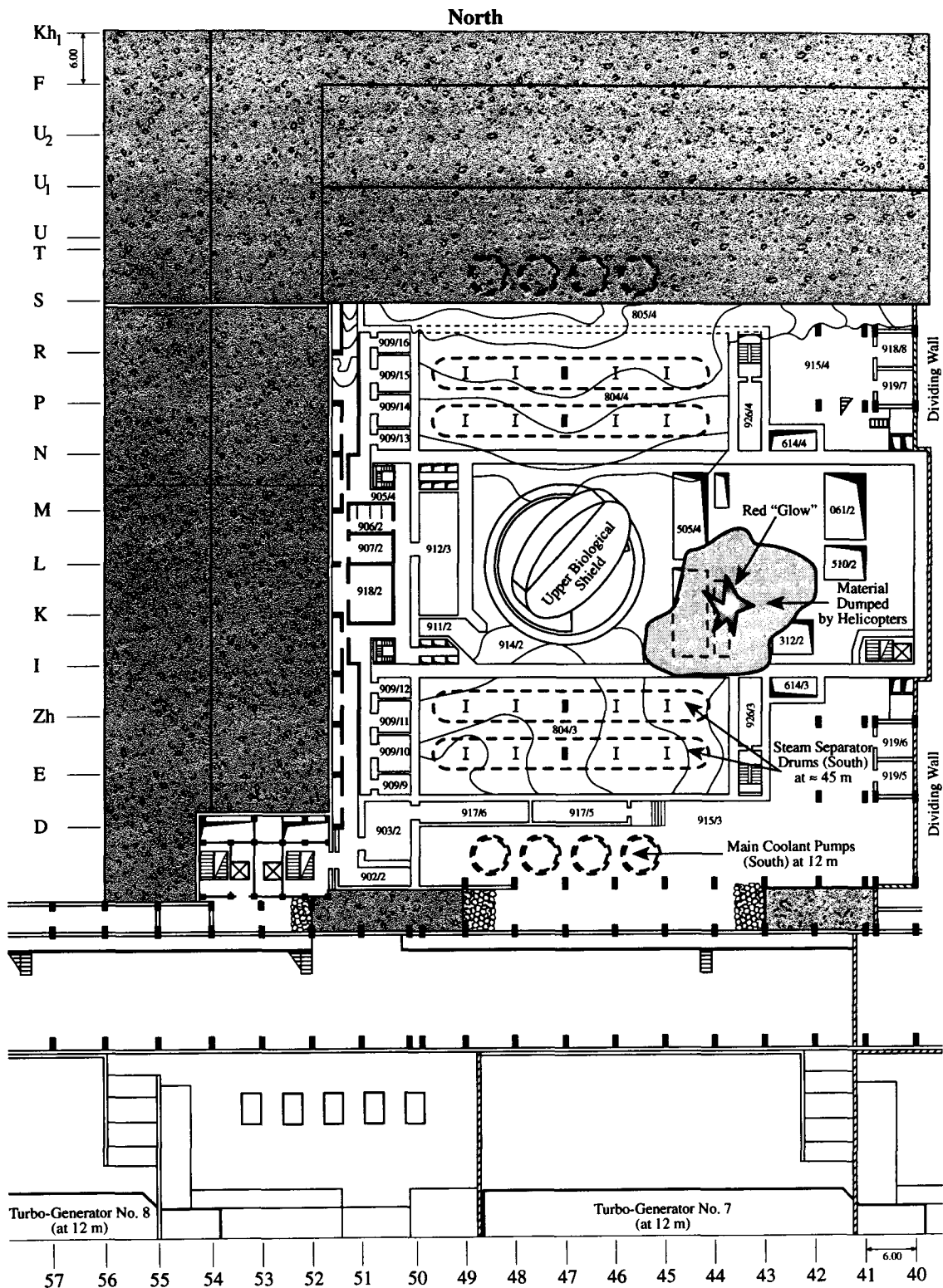


Fig. 2 Cross section (looking north) of the damaged Chernobyl Unit 4 reactor building and sarcophagus.

"liquid nitrogen was pumped. . . into all the spaces around the reactor vault" and "a cloud of cold nitrogen [gas] rose around the reactor. This proved to be an effective remedy. The temperature of the lower part of the reactor began to fall. At the same time, the air drawn by the chimney effect through the reactor core was gradually replaced by nitrogen, which suppressed the graphite fire. . . The fire began to die out."³⁷ A number of other sources also credit this action with having fulfilled its intended purpose (Ref. 16, p. 39; Ref. 23, pp. 88 and 89)^{38,39} (see endnote z, p. 19). Unfortunately, as with the material dumped to smother the core, this scenario does

not fully reflect the reality of events during the Active Phase.

By all indications,⁴⁰ the original intent of the May 1 decision was to pump liquid nitrogen into the core shaft to cool the core and to limit oxidation.⁴¹ An order went out to bring as much liquid nitrogen as possible to the station; and by May 6 (at 1:00 a.m., almost exactly 10 days after the accident and just after the end of the Active Phase) (Ref. 11, p. 140) the first tanker trucks were arriving on the scene. Two N₂ and O₂ makeup stations, to service the needs of Units 1 and 2 and Units 3 and 4 separately, were located on the territory of the power



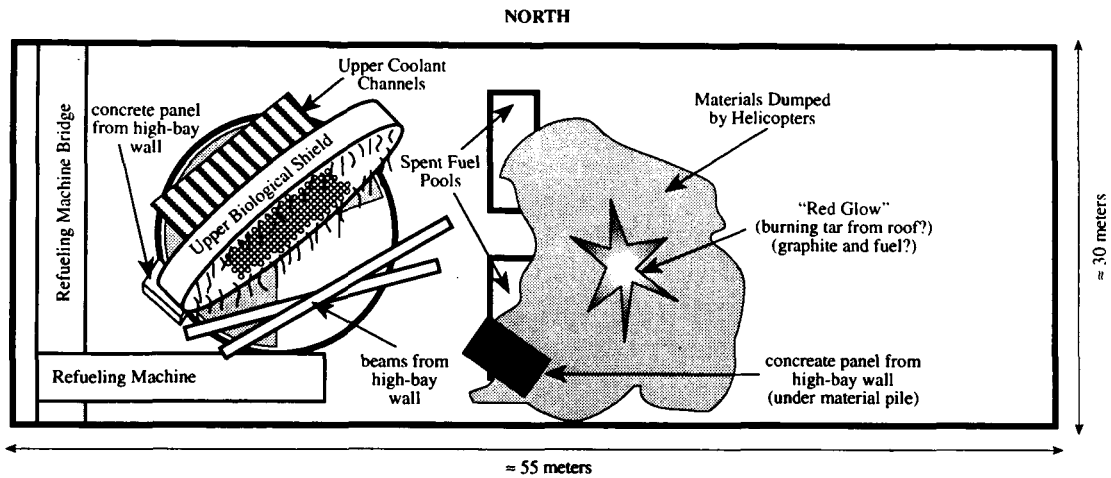


Fig. 4 Schematic of the central hall showing the location of materials dropped from helicopters.

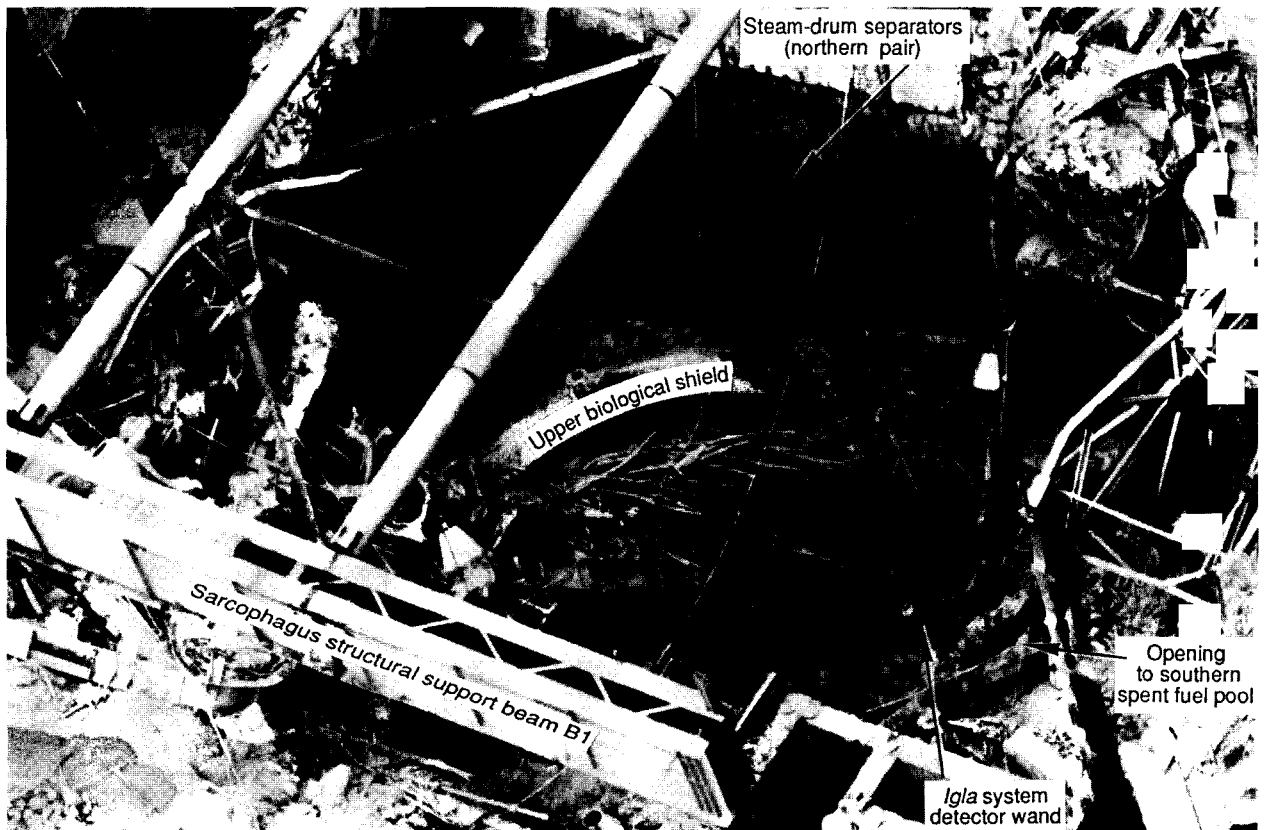


Fig. 5 Photograph of the Upper Biological Shield (UBS) viewed from the south-by-southeast taken in September 1986. (Pressure channels along the periphery of the UBS remain, whereas those in the middle have clearly been severed flush with the surface.) Note the absence of any bags of materials covering the UBS and surrounding structural debris. Note also the gaping hole in front of the UBS leading downward into the core area as well as the hole over the southern spent-fuel pool. (Photo courtesy of Valentin Ivanovich Obodzinsky of the Kurchatov Institute, Moscow.)

plant along with a compressor station just to the northeast of Unit 1 (Fig. 6). In addition, several large N_2 storage tanks were located close to the western wall of Unit 4, but because of debris from the explosions and high radiation fields in that area, this supply was unusable. The objective was to take advantage of the $500\text{-m}^3/\text{h}$ -capacity compressors to pump *gaseous* nitrogen approximately 400 m along existing 150-mm piping to the east face of Unit 3 (see endnote aa, p. 20). At this point, several walls (starting from the east side of Unit 3 and continuing through a number of rooms and Corridor 001 at level -1.0 m) were

to be drilled through to assemble another 200 m of piping that would carry nitrogen farther along until just before the western wall of Unit 4. Here the pipes would be directed upward to the second floor of the pressure-suppression pool into which the nitrogen was to be released.

The decision was made to pump in the hope that the *gaseous* nitrogen would eventually snake its way through steam distribution piping on the second floor of the pressure-suppression pool, through the steam distribution corridor, up through the subreactor region, and into the core to displace the oxygen and thus quench the fire (see Fig. 7

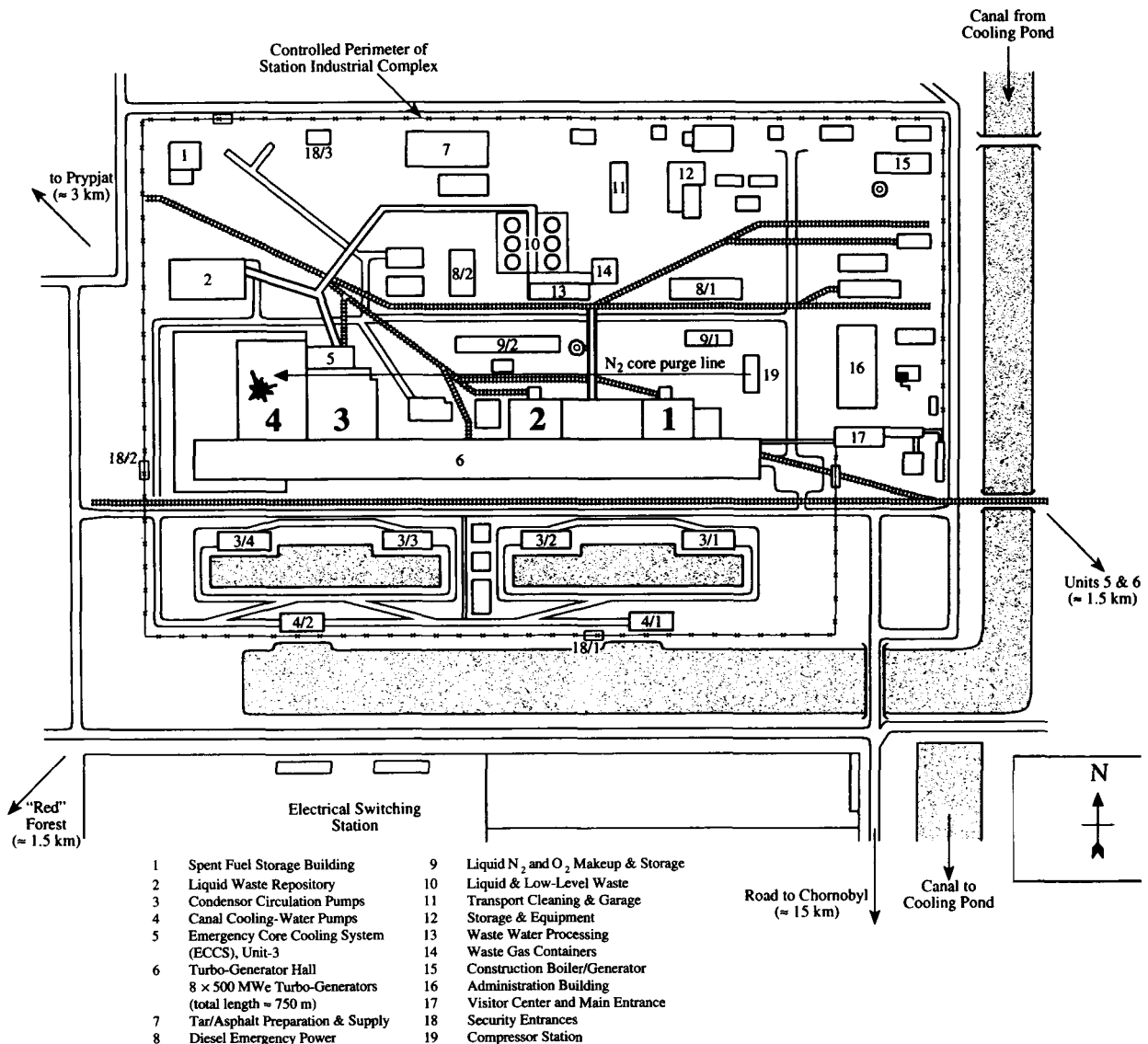


Fig. 6 Map of the Chernobyl Nuclear Power Station.

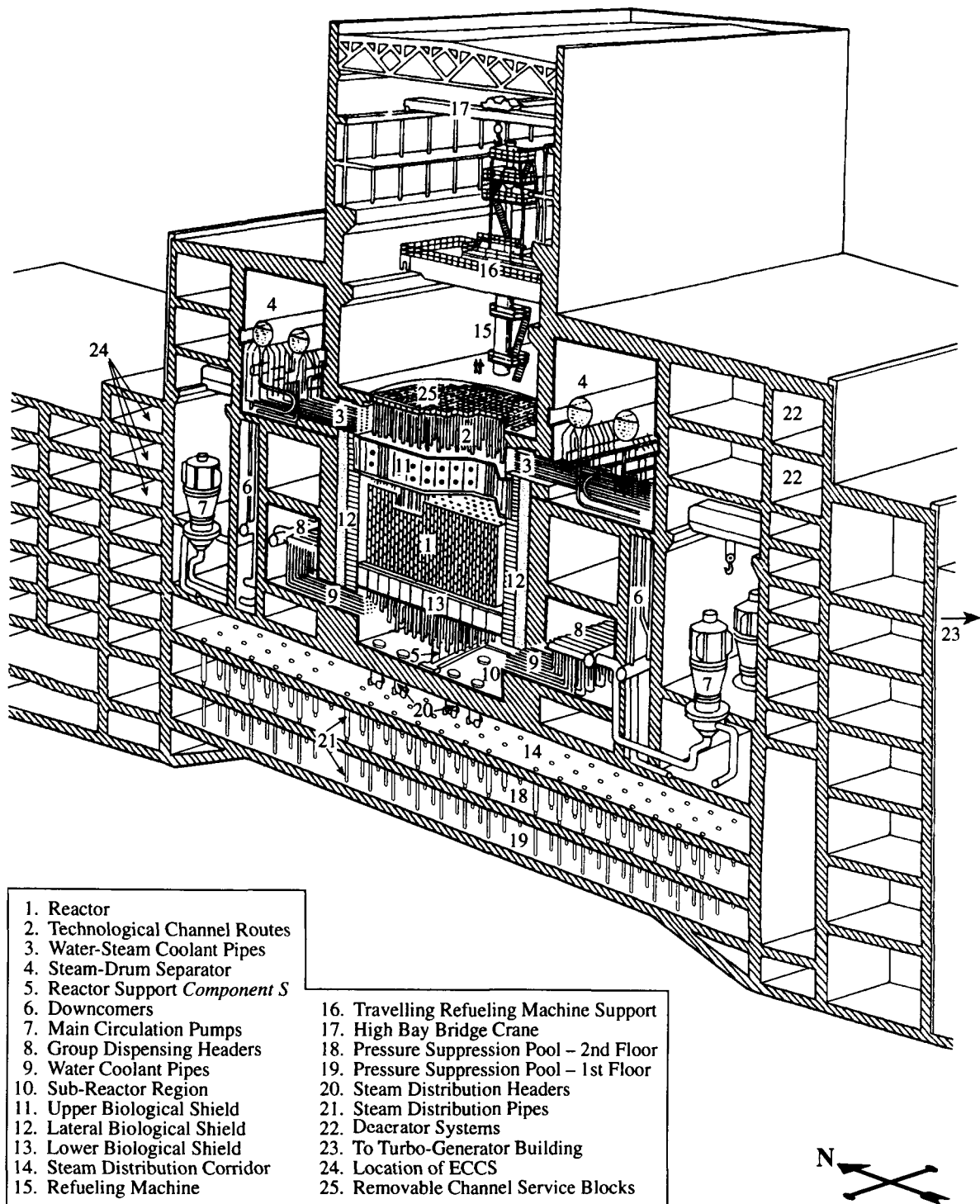


Fig. 7 Sectional view of Chernobyl Unit 4.

for a visual reference). Presumably this operation was carried out. The pumping itself, however, was not started before 1:00 a.m. on May 6 (i.e., after the Active Phase when, for all intents and purposes, the major part of the crisis was over). In fact, "Little of it [liquid nitrogen] was used, however, because after twenty-four hours even Legasov realized that the operation was a waste of time" (Ref. 11, p. 140). By this time the core had melted through the LBS and flowed into the lower regions of the reactor building where it "froze" (Fig. 8). Of course, at that point the nitrogen would have had no effect on the core. Moreover, a tremendous amount of nitrogen was needed considering that the free volume of air for the pressure-suppression pool alone under nominal condi-

tions is 3700 m^3 —this is especially true given the fact that the N_2/O_2 : $28/32 = 0.875$ mass ratio is quite inefficient for oxygen displacement (Ref. 40, pp. 2-46). In addition, later investigations have shown that, in addition to the very heavy damage suffered by areas immediately adjacent to the reactor that would permit nitrogen gas to filter out of the building and thus further reduce the effectiveness of the core purge, the LBS (weighing approximately 1200 tonnes-equivalent) had descended 4 m, crushed all the piping beneath it, and was heavily damaged itself because fuel had melted through it.⁴² Apparently, therefore, any attempt to pump nitrogen through the structural debris as suggested in the Soviet report would have proved futile (see endnote *bb*, p. 20).

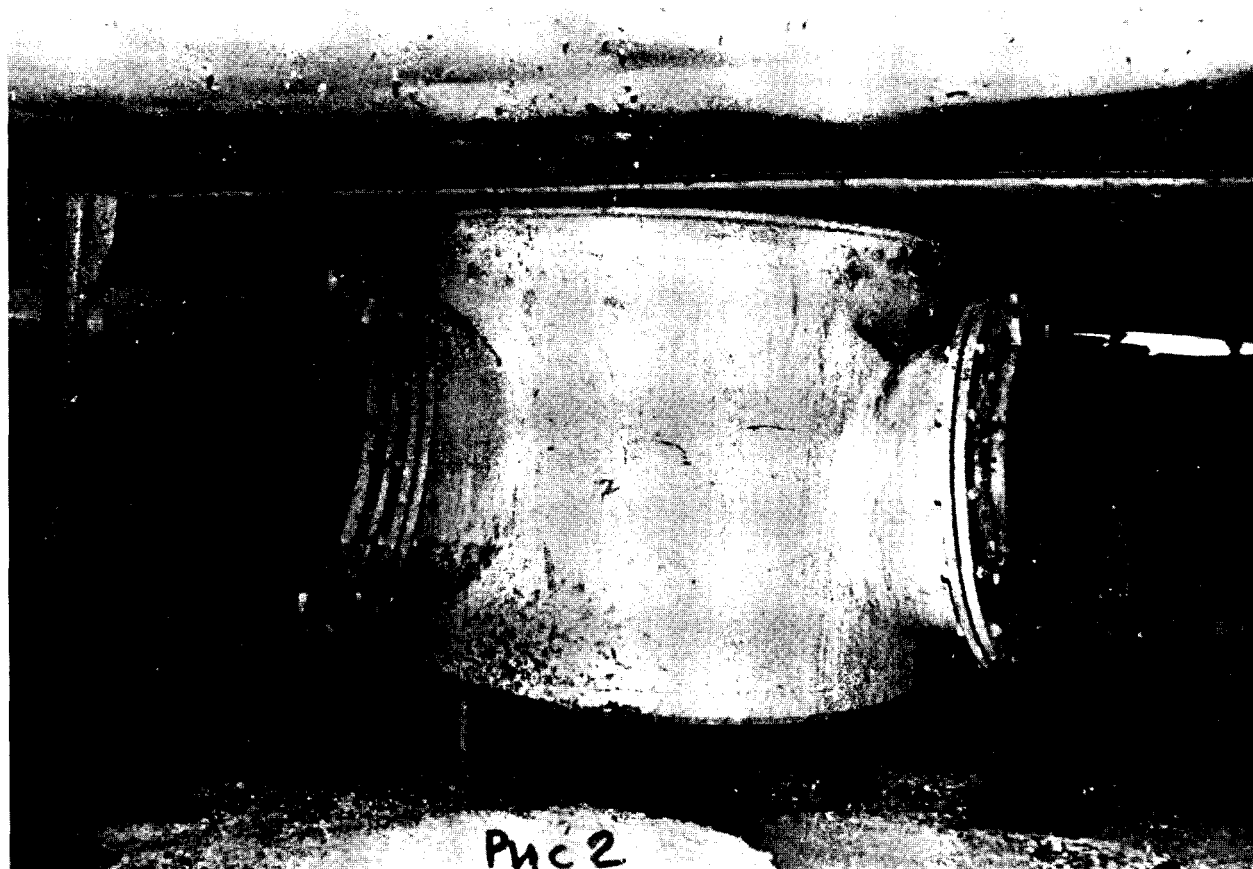


Fig. 8 Photograph showing one of the three Steam Distribution Headers (part of the Accident Pressure Relief System) with waterfall-like formations of frozen corium. This seems to indicate that, by the time the corium had melted through the Lower Biological Shield and reached the headers, it was in a very viscous, molasses-like state able to solidify very rapidly. (Note that at more than 1000°C the yield strength of stainless steel is significantly lowered and that its melting point is in the range of 1300 to 1700°C .) Note also that the header (as well as other structural components of the lower regions of the reactor building) suffered relatively little damage from the corium. (Particularly intriguing is the chain hanging from the header's left outlet together with the corium flow.)

Why the Soviets reported in Vienna that this method was a major contributor to successfully *cooling* the core and arresting the release of radioisotopes without positive confirmation is unknown.

Water Release from the Pressure-Suppression Pool (see endnote *cc*, p. 20). Soviet engineers became aware that the approximately 5000 tonnes of material thrown into the heavily damaged Central Hall threatened to compromise the structural integrity of the reactor support (see endnote *dd*, p. 20) and possibly push the molten core into the lower regions of the reactor building—creating, as it were, an *assisted* “China Syndrome” situation. It was reasoned that, if the molten fuel were able to penetrate into the lower regions of the reactor building, where the pressure-suppression pool and its 3200 m³ of water were located (see endnote *ee*, p. 20), the interaction of the fuel with water would generate a tremendous amount of steam that, in turn, would lead to enhanced releases of radionuclides into the environment (see endnote *ff*, p. 20). (Actually, “passages and air vents of the basement” also became flooded with water from broken auxiliary feed pipes used to supply water, albeit unsuccessfully, to the destroyed reactor during the first day after the accident.) (see endnote *gg*, p. 20). During the night of May 3 to 4⁴¹ (see endnote *hh*, p. 20), sometime before excavation design work for the flat-bed heat exchanger began (see endnote *ii*, p. 20), three volunteers (A. Ananenkov, V. Besspalov, and B. Baranov) (Ref. 4, p. 157) in diving suits managed to open slide or gate valves while fire brigade pumps were used to drain the pool. Medvedev states, “It took until May 8th [2 to 3 days after the Active Phase] for the task to be completed, by which time some 20 000 tonnes of highly radioactive water had been pumped out” (Ref. 37, pp. 58 and 59). The water was first pumped into Corridor 001 on the northern side of the basement of the reactor building and was later transferred to a liquid waste repository on the station grounds.⁴⁰ Not all the water was drained from the basement, however, and “fresh” water, in the form of precipitation, constantly enters the basement. According to a 1992 Soviet report, approximately 600 m³ of radioactive water (a considerable amount that today remains a major issue in plans for the construction of a new sarcophagus) is still located within the sarcophagus (Ref. 30, p. 42). Also, the lava that did encounter whatever water was left in the pressure-suppression pool turned into a hard, pumice-like substance—pieces of which actually floated away from the main pile and spread contamination throughout these lower areas.

Subfoundation Flat-Bed Heat Exchanger. Still another action taken to avert a possible “China Syndrome”

was the construction of a makeshift flat-bed heat exchanger (initially designed to employ liquid nitrogen as the working fluid) (Ref. 39, pp. 7-20) underneath the foundation of the Unit 4 building. It was reasoned that the planned 25 tonnes/day of liquid nitrogen would keep the soil frozen at a temperature of -100 °C and thus provide not only a stronger support for the building's foundation but also cooling the foundation with the hope of arresting the interaction of molten core with concrete. The decision to begin construction was made early on in the accident when little was known about the condition of the core or how extensively damaged the lower regions of the reactor building were, and construction was supposedly completed in late June. Table 3 provides a design schedule for the planned version of the heat exchanger.

By all accounts the heat exchanger was built, but its working fluid was water, and it was not completed until the end of June, approximately 8 weeks after the end of the Active Phase.⁴⁰ The initial design called for the entrance to an access tunnel to be located on the north side of the spent-fuel storage building (Fig. 8) where the construction and mining crews would be partially shielded from the most contaminated areas surrounding the destroyed Unit 4. This idea was quickly abandoned, partly because the spent-fuel storage building was located too far away (it would take too long to reach the intended mark below the core) and partly because the foundation had a vault-like shape running east to west; that is, the tunnel from the north would have to be deeper and not completely horizontal. Construction on the access tunnel was started about 2 to 3 days *after* the end of the Active Phase from the east side of Unit 3, whereas work on the heat exchanger itself began on June 3 and was completed on June 28 (Ref. 22, p. 97).

While design work was being completed on the heat exchanger, the 168-m access tunnel was excavated. A 2-m-diameter tunnel was dug with ribbed ceiling supports; its working diameter was approximately 1.8 m (Fig. 9). After the tunnel was completed, excavation continued for another 30 m to the west, after which the tunnel split to the north and to the south into 1.5-m-diameter “arms,” each extending 15.4 m. The height of the shafts was extended to 2.5 m where the top ran flush with the bottom of the foundation of the reactor building. When this set of “arms” was excavated, 100-mm-diameter coolant tubing sheathed with a thin plate of graphite was assembled and installed along with reinforcing iron rods, 25 mm in diameter, running horizontally and vertically and separated by a distance of 100 mm. Heat detectors were placed flush against the concrete foundation of the building as well as in other locations,

Table 3 Design and Construction Schedule for the Chernobyl Unit 4 Reactor Building Subfoundation Flat-Bed Heat Exchanger (Ref. 22, p. 338)

Goal/task	Organization(s) responsible	Scheduled completion
Heat exchanger plate	(1) PO-VNIPIET ^a	May 15
Feasibility study and validation	(2) Kurchatov Institute of Atomic Energy	
Borehole soil freezing design	(1) USSR Ministry of Transport Systems Construction	May 20
Horizontal tunnel design	(1) USSR Ministry of the Coal Industry	May 20
Heat exchanger plate design	(1) Atomic Power Design and Construction (2) USSR Ministry of the Coal Industry (3) Power Systems Construction (4) Hydro-Station Special Design	May 20
Cooling water supply system design	(1) PO-VNIPIET (2) Kurchatov Institute of Atomic Energy (3) Hydro-Station Design	May 25
Complete design implementation and appraisal	(1) Atomic Power Design and Construction (2) USSR Ministry of Coal Industry Equipment (3) Hydro-Station Special Design (4) Hydro-Station Design	May 30
Final recommendation for exploitation	(1) PO-VNIPIET (2) Hydro-Station Design	May 30

^aPO-VNIPIET—The Industrial Association—(All Union) Scientific Research and Design Institute of Power Engineering; Vladimir Aleksandrovich Kurnosov, Director.

and concrete was poured in to strengthen and protect the plate and to act as a major barrier to corium if it breached the foundation of the building. Moving back to the east, other sets of arms were thus excavated for the next sections of the heat exchanger. After the 30-m by 30.8-m by 2.5-m cavern and heat exchanger were completed, more concrete was poured to make the plate a massive integral object (Ref. 22, p. 338). After the tunnel had been sealed and pressure and leakage tests were conducted with the water working fluid, the project was abandoned.⁴⁰ Although a commendable attempt, with hindsight it proved ineffective because, as mentioned earlier, later investigations revealed that large quantities of corium cooled quickly in the lower regions of the reactor building approximately 9 to 10 days after the accident and thus caused relatively little damage to piping and other structural materials.

REASSESSMENT OF ACTIVE PHASE AMAs

The AMAs taken at Chernobyl during the first few days following the accident were generally ineffective.

The first attempt to supply water to the core from emergency auxiliary feed pumps to quench the core debris was apparently unsuccessful and quickly abandoned (Ref. 15, p. 40). The subsequent steps, namely (1) dumping of various materials into the reactor building to smother the fire, (2) supplying nitrogen to bring down the temperature in the core space and to reduce the oxygen concentration in the air, (3) construction of a flat heat exchanger beneath the foundations of the reactor building, and (4) release of water from the pressure-suppression pool, either were not fully implemented as reported (as in the case of nitrogen purging), failed to cover the reactor core (in the case of dumped material) apparently because the main goal was to stop the burning (which was visible) and not to cover the core (which was partially hidden from sight by piping and debris), or were not completed during the Active Phase of the accident (in the case of the makeshift flat heat exchanger beneath the reactor building, which was not completed until the end of June 1986). Moreover, once the corium-lava ate through the LBS, it had lost most of its heat energy in the melting process, and the decay heat had decreased significantly. In addition, the solidus of the corium-LBS mixture had increased (because the materials of the LBS

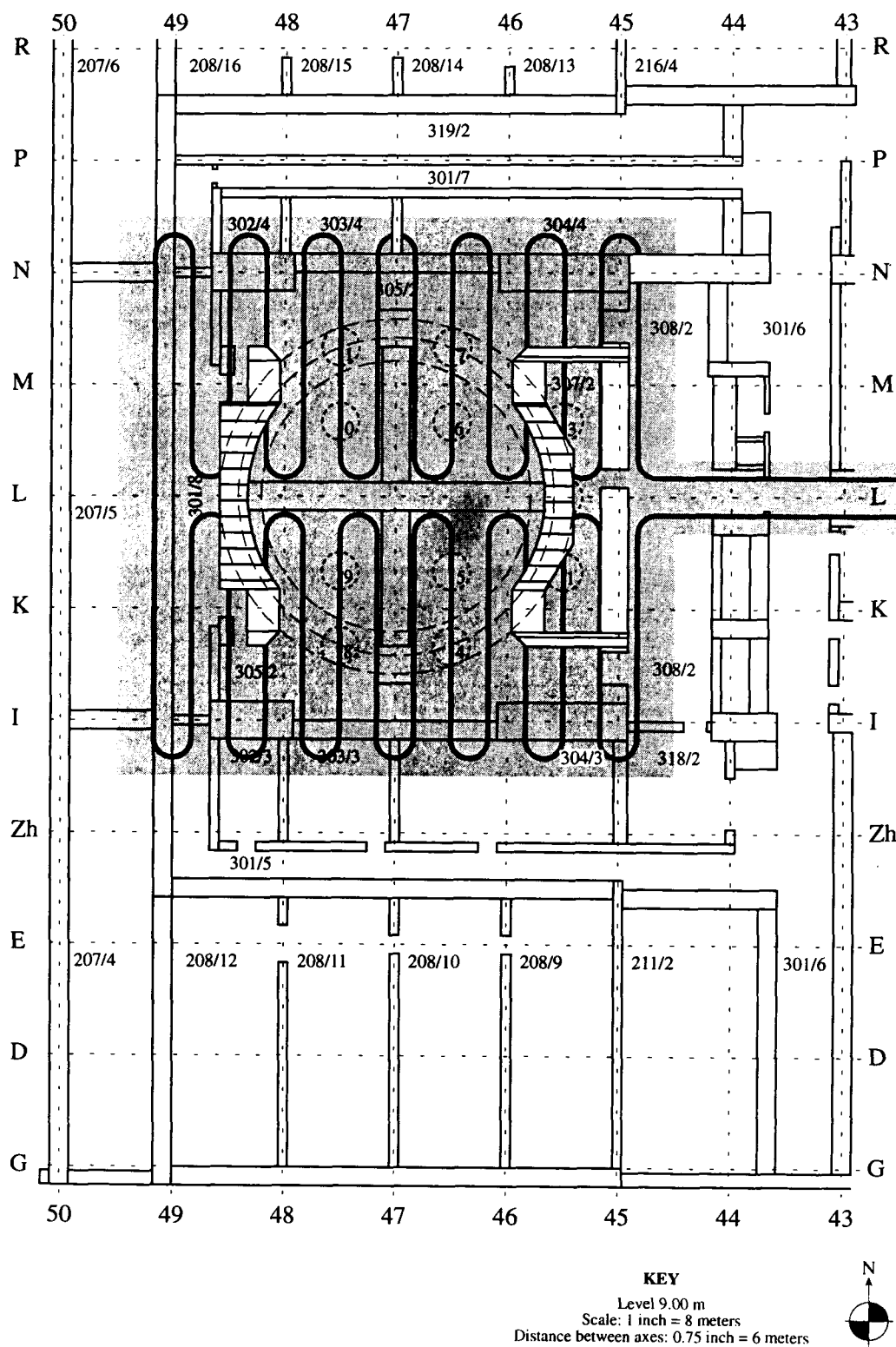


Fig. 9 Top view schematic of Chernobyl Unit 4 subfoundation flat-bed heat exchanger. Level 9.00 m: subreactor region.

and surrounding structures had complicated the chemistry) and therefore decreased further its ability to interact significantly with surrounding materials (see endnote *jj*, p. 21). The lava then flowed downward (along existing piping of the pressure-relief system and openings in the damaged building structures caused by the explosions)

and eventually into the pressure-suppression pool—all along the way solidifying in midflow, producing interesting stalactite-like formations and clogging piping but causing relatively little damage. A summary of AMAs undertaken during the Active Phase is shown in Table 4 and Fig. 10.

Table 4 Summary of Accident Management Actions Taken During the Active Phase of the Chernobyl Accident

Date	Time	Days	Action and reasoning
Saturday, April 26	≈01	0	Accident—destruction of Unit 4 reactor
Saturday, April 26	01:30–06:30	0–0.25	Firefighters extinguish blazes started by ejected core materials—mainly on the roofs of the auxiliary buildings, Unit 3, and the turbine-generator hall
Saturday, April 26	Morning	0.25–0.25	Feedwater pumps turned off—found ineffective apparently because piping to core destroyed
Saturday, April 26	>15:00	>0.75	Official recognition of what had happened and extent of damage, decision taken to begin material “bombing” campaigns
Sunday, April 27	≈10:00	≈1.4	Start of material “bombing” campaigns
Sunday, April 27	At night	≈2	Arrival of boron carbide
Wednesday, April 30	During day	≈4.5	Arrival of lead
April 30–May 1		4–6	Apparent ebb in releases followed by increase in releases to the end of the Active Phase
Thursday, May 1	During day	≈5.5	Decisions made to set up system to “purge the core with [ostensibly with <i>liquid</i>] nitrogen” and to design a subfoundation heat exchanger to mitigate against a possible “China Syndrome”
Friday, May 2	During day	≈6.5	Temporary halt to material bombing until after the Active Phase—feared that building support structures would be compromised that could initiate an “assisted” China Syndrome or steam explosion if reactor core were driven into the pressure-suppression pool
May 3–4	At night	≈8	Divers open slide valves to drain water from the pressure-suppression pool
Monday, May 5		9–10	Nitrogen-purging system installed
Monday, May 5		≈10	Peak of releases observed followed by rapid and significant drop
Tuesday, May 6	≈01:00	≈10	First tanker trucks arriving with liquid nitrogen. Soon afterward gaseous nitrogen started to be pumped into second floor of the pressure-suppression pool
Tuesday, May 6		10–11	Releases drop to more than three orders of magnitude less than during the initial 10 days—this signals the end of the Active Phase
Thursday, May 8		≈13	20 000 tonnes of highly radioactive water from the pressure-suppression pools had been pumped out and into Corridor 001
Tuesday, June 3		≈28	Construction start for subfoundation, flat-bed heat exchanger
Saturday, June 28		≈64	Completion of construction and testing of subfoundation, flat-bed heat exchanger

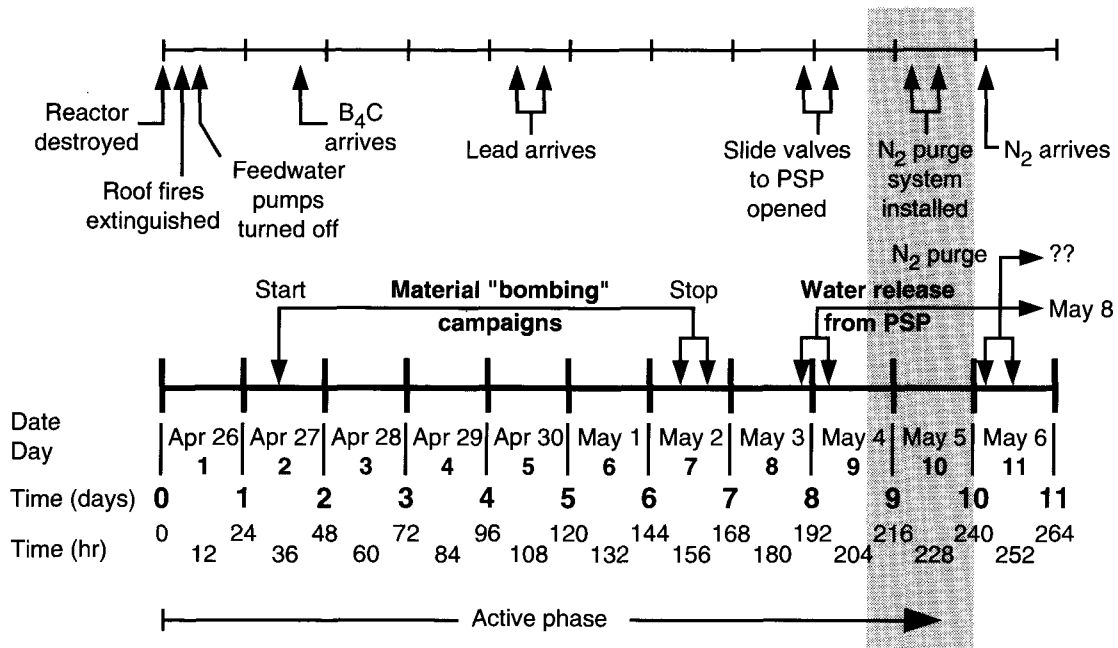


Fig. 10 Summary of accident management actions.

One interesting consequence of the fact that the corium-LBS did little damage to surrounding structures in the lower regions of the reactor building (especially considering the AMAs were ineffective and the burning-melted core fell into a more stable state with no human intervention) is that a "China Syndrome" now seems much less likely even for an accident as severe as Chernobyl. This has important implications for western safety concerns; namely, it may be possible to avoid a China Syndrome altogether at western nuclear power installations by designing a relatively simple, low-volatility material bed to "catch" ejected corium in the highly unlikely event of a pressure-vessel failure and let it interact with the bed material to form a quickly solidifying mixture. [Note that a typical 1000-MW(e) western pressurized-water reactor or boiling-water reactor has a UO₂ fuel load of approximately 75 or 115 tonnes, respectively, compared to the RBMK-1000 with a 190.3-tonne UO₂ fuel load; in the case of the Chernobyl accident, about 135 tonnes flowed into the lower regions of the reactor building.] If designed properly and combined with the western practice of flooding the corium with as much water as possible, the corium would be expected to bind chemically into a more stable state and thus reduce the pressure and contamination burden on the containment building.⁴³⁻⁴⁵

On a more somber note, the results of these investigations seem to support earlier western far-field source-term estimates that indicated that significantly more

volatile radionuclides were released into the environment than were reported by the Soviets at Vienna in August 1986. If indeed a major portion of the core were exposed virtually uncovered for approximately 9 days, it would have released more radioactivity into the environment than had been previously thought. The significantly increased incidences of children's thyroid cancers now occurring in Ukraine and Belarus may be evidence for such larger releases. One would wish that the current Russian government will soon release data and other information contained in the files and records of the former Soviet Governmental Commission on Chernobyl.

At the Vienna IAEA conference in August 1986, the Soviets stated that they had to choose one of two options to mitigate the releases: (1) Either "localize the focus of the accident by filling the reactor shaft with heat discharging and filtering materials" or (2) "Allow combustion processes in the reactor shaft to end naturally" (Ref. 15, p. 40). It now appears evident that the burning of the core and the release of radionuclides that took place during the Active Phase of the accident in fact at least partly stabilized themselves rather than as the result of Soviet actions. From the point of view of the safety of personnel assigned to implement AMAs during the Active Phase, the Soviet claim that "...decisions made [to mitigate the consequences of the accident] were primarily the right ones" (Ref. 15, p. 41) seems correct. However, the actions required to consummate them were not always properly carried out.

Given this new information, western assessments of the effectiveness of AMAs must also be reassessed;⁴⁷ for example, the IAEA's INSAG-1 report states

The accident management actions taken at Chernobyl were, generally, quite successful... [the] dumping [of] materials into the reactor well..., supplying nitrogen to bring down the temperature in the core space and to reduce the oxygen concentration, and the construction of a flat heat exchanger beneath the foundations of the reactor building, stabilized the situation at an early stage (about nine days after the initiating power surge) of the accident" [emphases added] (Ref. 16, p. 43) (see endnote *kk*, p. 21).

This indicates that western experts, no less than their Soviet counterparts, at least partly misjudged the efficacies of the remedial measures.

NOTES

^aIn the INSAG-7 report published January 1993, the International Atomic Energy Agency's (IAEA's) International Nuclear Safety Advisory Group identified the root cause of the Chernobyl accident as the RBMK reactor design and thus shifted the blame from the operators to the designers. (Their original report, written on the basis of partial information supplied by Soviet officials a few months after the accident, laid a large part of the blame on the operators.) It is also clear that the design faults identified in the report are associated with features of the RBMK that were intended to make it a more efficient reactor. The INSAG report makes clear that the problems with the RBMK plant design had, in fact, been recognized before the Chernobyl accident.

Pravda (15 July 1986) reported that the director and the chief engineer of the Chernobyl power station were dismissed because they were "unable to ensure the correct firm leadership and proper discipline." It continued by saying that the two officials "showed irresponsibility and inefficiency. They were unable to assess what had happened or to take measures to organize efficient work by all subunits following the accident." The article also said that Alexander Sicharenko, an engineer and local party official from Prypjat, and another party official had ignored their duties during the evacuation of employees and residents. The latter was stripped of his Communist Party membership, and the former was given a "severe reprimand."

^bThere is general agreement among cancer researchers that few cancers will appear before a "latent" period. That period is 5 years for leukemias, 15 years for some other tumors, but may be as little as 2 years for childhood leukemias and thyroid cancers.

^cFor a more extensive treatment of these cover-ups, see the sections *Coverup and Controversy* and *The Wages of Fear*, pp. 150–154 in *Ecocide in the USSR: Health and Nature Under Siege*, Murray Feshbach and Alfred Friendly, Jr., Basic Books, Inc., New York, 1992.

^dI have spoken with a number of Russian and Ukrainian scientists over the past 3 years who assure me there is unreleased information on almost every aspect of the accident,

most of which forms the bulk of the as yet inaccessible Soviet Government Commission's archives on the accident.

^eThe first hint that the Soviets were not completely forthright at the meeting of experts in Vienna in August 1986 was Academician Valerij Legasov's statement before the Soviet Academy of Sciences in October 1986, "I did not lie at Vienna, but I did not tell the whole truth." Mallinckrodt Professor of Physics, Dr. Richard Wilson of Harvard University, was quoted this by Academician Andrei Sakharov in February 1987, confirming what several experts had been cautiously wary of at the Vienna meeting. Even more unfortunate, under the pretext of avoiding mass hysteria and "radiophobia" in the wake of the accident, it was clear from the Soviet press that the central authorities told even less to their own citizens; for example, it was not until February 1989 that the republican governments of Belarus and Ukraine published maps of radiation fallout, followed hastily by the all-union authorities publishing a similar map in *Pravda*. Moreover, in the November 1990 issue of *Pryroda* (Nature), Kalugin wrote that "The members of the Soviet delegation were strictly instructed not to meet with foreigners, not to answer any questions on their part, and to follow the published word in every respect. Only because of the resolute stand taken by [Dr. Valerij] Legasov was it possible to go away from this policy."

^fIt is interesting that the word chosen in Russian and Ukrainian instead of mitigation is the stronger term "liquidation." To this day those who were or are engaged in clean-up efforts at Chernobyl are officially called "liquidators."

^gSee Ref. 24, pp. 121, 174–178.

^hSee Ref. 11, p. 84. This and subsequent information are very important details that establish the fact that any *early* reconnaissance of the destroyed Unit 4 (let alone proper air sampling above the reactor) would have been impossible until the Government Commission approved such actions. Commission members did not leave on a flight from Moscow to Kiev until 10:00 a.m. (p. 95), while Valerij Legasov, Academician, First Deputy Director of the Kurchatov Institute of Atomic Energy, and head of the Soviet delegation to Vienna, did not leave Moscow until 4:00 p.m. (p. 96).

ⁱEven after the operators had been told how extensive the damage was, they didn't believe the firefighters who had been on the roof to look down into the Central Hall (see Ref. 23, p. 84).

^jBoris Shcherbina (Soviet Deputy Prime Minister and 1st Chairman of the Government Commission on Chernobyl) and Valerij Legasov did not arrive in Prypjat until 8:00 p.m. the day of the accident (see Ref. 11, p. 101).

^k"[the team] had flown over the damaged reactor at a height of eight hundred feet in a helicopter belonging to the civil defense. . . . The explosion had destroyed the reactor and ignited the graphite in its core. The Upper Biological Shield, weighing one thousand tons, had been blown to one side, leaving the inside of the reactor open to the sky. It was red-hot from the graphite fire" (see Ref. 11, p. 101).

^lActually it wasn't the "concrete structures supporting the reactor" that the Soviets were worried about but rather the integrity of the 5.3-m-tall steel cross-shaped reactor support,

Component S. This seems to be confirmed by "The U.S.S.R. team reported that they were concerned that any further additions of material could overload the *structural support of the reactor vault* (see Ref. 17, p. 33).

^mAlso in Ref. 23, p. 39, it is expressly stated without substantiation that "These phenomena [the alleged behavior of releases as shown by the bathtub curve, and especially the final sharp drop in releases] were results of specially adopted measures [Accident Management Actions] that led to the binding of fission products into more stable chemical forms."

ⁿ"The [accident management] measures adopted by the Governmental Commission significantly diminished direct contact of the damaged [!] reactor core with the atmosphere and prevented the spread of the melted nuclear fuel beyond the bounds of Unit 4. The over 5000 tons of various materials thrown into the reactor created the needed insulating plug over the reactor." (Emphasis added, translation from Ref. 22, p. 37.)

^oIn architectural-engineering terminology, the Soviets typically use the term "scheme" to indicate a major component of a system. In this case the Lower Biological Shield is often referred to as *Scheme OR*, whereas the Upper Biological Shield is referred to as *Scheme E* (hence its nickname "Elena"). For clarity, the term *Component* will be substituted for *Scheme*; so, for example, the Lower Biological Shield will be referred to as *Component OR*.

^pGiven the fact that the roof of Unit 4 was covered with a large amount of asphalt-like (bitumen) material, an argument has been put forth that it wasn't graphite at all that was seen burning—producing the red glow—but rather a large quantity of asphalt that collapsed into the Central Hall together with the roof. Unfortunately, this is impossible to verify because the Central Hall area, and especially the pile of debris over the spent-fuel pools, has not yet been investigated.

^qSerhij Vasylovich Shirokov (see Ref. 25) flew over the destroyed reactor building twice: on May 1 and on May 5 (on the 6th and 10th days following the accident). According to Shirokov, on May 1 it was quite difficult to see clearly into the area of the Central Hall because "thick, white" smoke was continuously billowing out, the unsteady motion of the helicopter made it difficult to get a fix on objects, and to avoid extreme exposures to the helicopter's crew the hover period was quite short. He asserts, however, that on May 1 the red glow in the vicinity of the southern spent-fuel pool was still visible, and a faint glow from the area of the reactor shaft could also be seen. Other smaller "glows" were visible through holes in the roof of the turbo-generator hall created by debris ejected during the explosion(s) that burned or melted through the roof. These "glows" were presumably chunks of core material that had not completely cooled.

^rAside from general air traffic control for the helicopters, it would have been impossible for the observer on the roof to guide the helicopter to the correct position above the reactor because he was over three kilometers away.

^sAir Force Major General Antoshkin was told, "...well after midnight on April 27th... everything depends on you and your helicopter pilots now, general. The crater has to be sealed

off tightly with sand, from above." By the time a suitable landing pad was established and enough men were rounded up to collect sand from the banks of the river into bags, the first bombing runs did not begin until sometime during the morning of April 27, or possibly one and one-half days after the accident. (Emphasis added, see Refs. 11, pp. 107-109, and 24, pp. 179 and 181.)

^tIt is unclear why, given this information, the IAEA in its *International Chernobyl Project* report continued to claim that the reactor had been covered with this material.

^uInterestingly, testimony given by V. Ya. Prushinsky, chief engineer of USSR Division of Nuclear Energy, states, "On May 4th, I flew over the reactor in a helicopter with Academician Velikhov. After studying the destroyed reactor building from the air, Velikhov admitted that he did not know how to bring the reactor under control. He sounded quite worried. And this was after 5,000 tons of assorted materials had been dumped into the gaping hole left by the nuclear blast" (see Ref. 24, pp. 182-183).

^vAs further evidence that the dumped material had not hit the intended target until it was too late, "Still, in the fading light of the evening [of May 6th], a last run by a helicopter over the reactor showed a small but bright spot of red in the crater. . . . The next day [May 7th] a further eighty tons of lead were dropped into the reactor, after which the luminescence ceased" (see Ref. 11, p. 141). From this account it is unclear what or where the "small but bright spot" was and contradicts radiochemical analyses (see upcoming article) that indicate essentially no lead is found in samples of solidified corium.

^w"On May 5, a cold nitrogen inlet system was located in the space under the reactor pit in order to provide additional cooling and to reduce the oxygen concentration" (see Ref. 28, p. 15). Although the purging system may have been installed, the nitrogen did not arrive until 1:00 a.m. on May 6—almost exactly 10 days after the accident—just after the end of the Active Phase (see Ref. 11, p. 140).

^xNote that here (as the primary source material) liquid N₂ is not explicitly mentioned, although in Ref. 22, p. 84, it is specifically stated that the reactor was cooled with *liquid* nitrogen. This is also the case in Ref. 16, p. 46, and the IAEA's *International Chernobyl Project* (Ref. 23, pp. 88 and 90) which references other sources, including INSAG-1.

^yIn what appears to be an indication of just how much emphasis was placed on *liquid* nitrogen as a working fluid, Valerij Legasov is alleged to have proposed, "Why not pump it [nitrogen] in to freeze the earth beneath the foundations, drive out the oxygen and smother the fire?" (see Ref. 11, p. 139).

^zRef. 23 states that (translation) "... members of the Governmental Commission decided to begin pumping *liquid* nitrogen into region underneath the reactor." Moreover, whereas the INSAG-1 report only hypothesizes a scenario for the Active Phase, The *International Chernobyl Project* report is more affirmative in its analysis, "In the next five days [April 27–May 1] the rate of release decreased to a minimum—approximately six times less than the initial release. During this stage of the accident, a *decrease in the release rate*

was achieved by measures taken to quench the burning graphite and by the cooling of the core [emphasis added]. These measures, including the dumping of 5000 tons of boron carbide, dolomite, clay, and lead into the core from helicopters provided filtration for the radioactive substances being released from the core." (p. 90). This appears to contradict a different conclusion reached on the previous page, "By May 6th the temperature of the [reactor] vault began to fall; the reasons remain unknown to this day. Possibly, this came about thanks to the action of the boron carbide and sand, and possibly due to the burning away of graphite or the further melting of the fuel which then took on a 'safer' [more stable?] configuration."

There are also other inconsistencies in this report; for example, on p. 88 we read (translation), "Part of the problem had to do with the fact that no plan had been prepared in advance in the event of an accident with such large-scale and prolonged releases of radioactive substances; for example, during the first few days after the accident, when the Governmental Commission was stationed in Prypjat, its members had neither respirators nor individual dosimeters. At the station itself, automatic means for external dosimetry control were lacking. . . ." Yet on the following page, "By the beginning of May, monitoring above [the reactor] began with the help of helicopters. By August [emphasis added], special detectors (diagnostic buoys) were able to be placed in the debris of the core close to the Upper Biological Shield. . . . These detectors measured gamma radiation, thermal conductivity, air temperature, and convection currents." The report continues, "By May 1st, the temperature of the core began to increase, which possibly arose as a result of fission product decay heat within the sealed [?!] reactor" [emphasis added].

Not only could the temperature of the core not be established to any degree of accuracy during the Active Phase but also even if the temperature could have been measured on May 1, it does not necessarily follow that the temperature was decreasing before that date, especially since the core was neither sealed nor covered. It appears likely that the temperature of the core, if not increasing during the first five days, was at least staying more or less constant.

^{aa}The 500-m³/h pumping rate appears low by approximately an order of magnitude. A typical U.S. liquid-nitrogen trailer truck can pump 650,000 ft³ to ambient in about 2 hours or at a rate of 9,200 m³/h. Given the 3700-m³ volume of the pressure-suppression pool, the volume of gaseous nitrogen would be sufficient. However, whether this would have displaced the oxygen in the area of the core is strongly dependent on the displacement efficiency, which, in turn, is strongly dependent on geometric factors. Because the core area and surrounding areas were heavily damaged and there was a strong updraft from the fires, without more information it is difficult to conclude that even gaseous nitrogen purging would have been effective, even if the pumping had started before the end of the Active Phase (see ahead).

^{bb}Interestingly, Taras Plokhij and Nikolai Steinberg, according to Piers Paul Reid, ". . . from the beginning, both thought this an absurd idea. If the explosion had been

contained within the structure, it might have made sense, but because the whole unit had been ruptured, all the nitrogen would escape into the open air" (see Ref. 11, p. 139).

^{cc}The pressure-suppression pool consists of two floors of water- and air-filled chambers with telescoping steam bubbler tubes (225 per floor) penetrating the slab between the two floors. The nominal water depth is 1.2 m, and the total water and air volumes on the two floors are approximately 3200 and 3700 m³, respectively. The pressure-suppression pool, as with the steam distribution corridor above it, is rated at a design pressure of 0.25 MPa (0.36 psig) (see Ref. 40, pp. 243-246).

^{dd}Again, it is unclear when the authorities realized that the material wasn't being thrown into the reactor shaft but rather off to the side to smother the "red glow." In addition to the influence fear may have had on judgment, one can understand why helicopter pilots and their crews (who were unfamiliar with the layout of the Central Hall and the location of the reactor shaft) were unable to differentiate the location of the "red glow" from the location of the reactor shaft. This is especially true given that smoke was billowing out from the Central Hall and debris from the explosion obscured visibility; however, if it is the case that reconnaissance teams of investigators and people managing the AMAs flew over the destroyed reactor during the Active Phase (assuming they were familiar with the layout of Unit 4), one wonders why the concern of "compromising the structural integrity of the reactor support" was raised.

^{ee}Interestingly, after a reconnaissance in the early hours of May 2, it was found that the 2nd (upper) floor of the pressure-suppression pool had no water, whereas the 1st (lower) floor contained only about 200 m³ of water (see Ref 11, p. 134).

^{ff}This is in contrast to western accident management principles which call for flooding a damaged reactor with as much water as possible until the situation is stabilized; i.e., the presence of great quantities of water is seen as beneficial to mitigating the consequences of severe accidents. Note, however, this assumes the presence of a containment building and filtration systems that would retain steam-generated releases. Chernobyl had no containment building, and water supply systems and the building itself were destroyed.

^{gg}Reference 11, pp. 134-137, provides an account of the draining of this area of the reactor building. This task was completed (after a 30-hour operation) by midnight May 7—2 days after the end of the Active Phase. Considering that the molten corium turned to a pumice-like substance only on the first floor of the pressure-suppression pool (see Chap. V of Ref. 11), the "draining of the basement" detailed here must have taken part in other regions of the vast basement of Unit 4.

^{hh}According to testimony by G. A. Shasharin, "On May 4th we found the gate valve which had to be opened in order to drain water from the lower part of the suppression pool. There was little water in it. We looked into the upper pool through the hole of the reserve passage and found it empty" (see Ref 24, p. 203).

ⁱⁱReference 40, pp. 7-20, reports that this occurred on the night of May 6-7 (i.e., after the Active Phase of the accident

was over), which confused the draining of the pressure-suppression pool with the draining of other regions of the basement that had become flooded as a result of the fruitless attempt to provide feedwater to the reactor in the earlier hours after the accident.

^{jj}Eutectic materials form readily during severe accidents and can lower melting or freezing temperatures by several hundred degrees.

^{kk}Another premature assessment of the accident is "The Chernobyl-4 accident generally was judged to identify no significant new lessons for the nuclear power industry outside the USSR."

REFERENCES

1. D. A. Powers, T. S. Kress, and M. W. Jankowski, "The Chernobyl Source Term," *Nucl. Saf.*, 28(1): 9 (January-March 1987).
2. In Belarus, for example, the incidence of thyroid cancers in children has increased from approximately two cases per year before Chernobyl to thirty cases in the first half of 1992 alone. More than half (71) of the 131 cases reported to mid-1992 since 1986 were in the Homel oblast (state) just north of the Kyjiv oblast in Ukraine where the Chernobyl Station is located. [Gimini Seneviratne, WHO Says Belarus Cancers May Stem From More Than Chernobyl, *Nucleonics Week*, 33(51):12 (December 17, 1992).
3. In another article, a group from the World Health Organization states that "The Belarus experience suggests that the consequences to the human thyroid, especially in fetuses and young children, of the carcinogenic effects of radioactive fallout are much greater than previously thought." [Ann MacLachlan, Rising Children's Thyroid Cancers Indicate Growing Chernobyl Link, *Nucleonics Week*, 33(37): 8 (September 10, 1992)].
4. Only one of several studies that contradict official Soviet figures presented at the Vienna Meeting of Experts is that of Edward A. Warman, *Soviet and Far-Field Radiation Measurements and an Inferred Source Term From Chernobyl*, Presented at the New York Chapter's Health Physics Society Symposium on the Effects of the Nuclear Reactor Accident at Chernobyl, April 3, 1987, Brookhaven National Laboratory. Warman concludes that "Approximately 30 to 50 percent of the available radiocesium and at least 40 to 60 percent of the available radioiodine appear to have been released to the atmosphere during the accident." This is in stark contrast to the officially reported releases of 10% Cs-134, 13% Cs-137, and 20% I-131 at the IAEA's Meeting of Experts in Vienna in August 1986.
5. See, for example, Chapter 5 "Safety in the Soviet Nuclear Power Industry" in David R. Marples, *Chernobyl & Nuclear Power in the USSR*, St. Martin's Press, New York, 1986; Maurice Strong, 40 Chernobyls Waiting to Happen, *The New York Times*, CXL1 (48913):15 (Sunday, March 22, 1992); and Chernobyls-in-Waiting, *The Economist*, 324(7772):18 (August 15, 1992). Moreover, INSAG-7 concludes that the Chernobyl accident stemmed from a deficient safety culture—not only at the Chernobyl Station but also throughout Soviet design, operating, and regulatory organizations for nuclear power that existed at that time.
6. Esther B. Fein, Soviets Curb News About Disasters, *The New York Times*, CXXXVIII(47,853):A15 (April 27, 1989).
7. Peter Gumbel, Villagers Suffering Chernobyl's Fallout Face Soviet Silence, *The Wall Street Journal*, CCXIII(44): A1 (March 6, 1989).
8. "New criminal charges concerning Chernobyl are being prepared by the Soviet Union's prosecutor general, Nikolai Trubin. In early February, Trubin alleged that the extent of the Chernobyl-4 accident in 1986 was covered up at the time, and also that officials have not sufficiently protected people from the effects of the released radioactive material for years since then." [Late News in Brief, *Nucl. News*, 99 (March 1991)].
9. Chrystia Freeland, Waiting for the Next Chernobyl, *The Financial Times of London*, No. 32,043 (Week 16):20 (April 21, 1993).
10. From a private conversation with Prof. Richard Wilson of Harvard University, July 1990. One year later, speaking at the 1st International Sakharov Conference on Peace, Progress, and Human Rights in Moscow (May 22, 1991), Prof. Wilson summarized several aspects of the accident's consequences subjected to secrecy by the Soviet central authorities:
 - About six pages concerning radioactivity released in Belarus were removed from the official report just prior to the August 1986 IAEA meeting and were not discussed.
 - Several pages detailing large quantities of radionuclides deposited 100 km and more northeast of Chernobyl in the Brjansk oblast of Russia were removed from the report following directives from the Soviet Central Committee.
 - Dosimeters in possession of physicians and private individuals were locked up by the KGB.
 - Publication of "unauthorized" measurements of radioactivity were forbidden—and even in 1990 still were forbidden (emphasis added).
 - Scientists who lectured on radioactivity to the public were forbidden to quote foreign sources of information.
 - Physicians in Ukraine and Belarus were forbidden to mention radiation in their diagnoses.
 - Appeals by private individuals in Belarus to children not to drink milk in the first weeks of May 1986 were stopped for fear they might cause panic.
 - All health records of the "liquidators" of the consequences of the accident (soldiers and others who worked cleaning up the 30-km exclusion zone and constructing the sarcophagus) are "missing." (Since the collapse of the USSR, these data have begun to slowly surface.)
11. Piers Paul Reid, *Ablaze*, p. 207, Random House, New York, 1993. The first point in this list is confirmed by Piers Paul Reid: "six pages covering the serious contamination of the area north of G[H]omel had been removed. [Prof. Richard] Wilson was later told that this had been done on Legasov's instructions."
12. Yevgenia Albats, The Big Lie: Who Will Answer for Hushing Up the True Causes and Consequences of the Chernobyl Tragedy? *Moskovskije Novosti*, pp. 8-9, October 15, 1989. See also N. Matukovskyj, Catastrophe: What the Lessons of Chernobyl Teach, *Izvestia*, NQ 86 (22989):3 (March 26, 1990).
13. Lyubov Kovalevska, Physicians in the Role of Attorneys for the Ministry of Atomic Energy, *Ukrainska Hazeta (Ukrainian Newspaper)*, No. 3 (April 1-7, 1993). Disclosed is a document entitled "About the Strengthening of the Regime of Secrecy During the Implementation of Task Regarding the Liquidation of the Consequences of the Accident at the ChNPS" [by order of

- the Head of the Main Administrative Division of the USSR Ministry for the Protection of Health, Ye. Shul'zhenko, (U-2617 "S") signed June 27, 1986].
14. Yaroshinskaya Alla, "Forty Secret Protocols of the Kremlin Wise Men: The Lie About Chernobyl is as Frightening as the Catastrophe Itself," *Izvestia*, 98 (23672):3 (April 24, 1992).
 15. U.S.S.R. State Committee on the Utilization of Atomic Energy (comp.), *The Accident at the Chernobyl Nuclear Power Plant and Its Consequences; Part 1: General Material*, report presented at the International Atomic Energy Agency Expert's Meeting, Vienna (August 25–29, 1986), p. 41. This is not to imply that during the Active Phase it was necessarily known that the core was uncovered. It appears this was a mistake and not known for some time—how much time is not clear. However, certainly by the August 1986 meeting, after Soviet RBMK experts would have had more than 3 months to examine directly the remains of the reactor building by flying over it in a helicopter (as well as the hundreds of photographs of the core shaft region), someone would have realized that the core had, in fact, not been covered.
 16. The International Nuclear Safety Advisory Group (INSAG-1), *Summary Report on the Post-Accident Review Meeting on the Chernobyl Accident*, August 30–September 5, 1986, International Atomic Energy Agency (IAEA), Vienna, Austria, Report GLC(SPL.I)/3 Safety Series No. 75-INSAG-1, September 24, 1986, p. 43. "The accident management actions taken at Chernobyl were, generally, quite successful... [the] dumping [of] materials into the reactor well..., supplying nitrogen to bring down the temperature in the core space and to reduce the oxygen concentration, and the construction of a flat heat exchanger beneath the foundations of the reactor building, stabilized the situation at an early stage (about nine days after the initiating power surge) of the accident."
 17. M. W. Jankowski, D. A. Powers, and T. S. Kress, "Onsite Response to the Accident at Chernobyl (Accident Management)," *Nucl. Saf.*, 28(1): 36–42 (January–March 1987).
 18. See also the 100% core-inventory radioiodine release estimate in an unpublished paper cited in D. A. Powers, Carburation as a Mechanism for the Release of Radionuclides During the Chernobyl Accident, in *Proceedings of the First International Workshop on Past Severe Accidents and Their Consequences, October 30–November 3, 1989*, p. 117, E. P. Velikhov and L. A. Bolshov (Eds.), Nauka, Moscow, Russia.
 19. See also a table summarizing release estimates from a number of countries on p. 612 in U. S. Nuclear Regulatory Commission (comp.), *Report on the Accident at the Chernobyl Nuclear Power Station*, Report NUREG-1250, U.S. Government Printing Office, Washington, DC, 1987. The widely ranging estimates are partially due to the generally unrefined and sketchy data presented in Vienna by the Soviets and the fact that most release data were suppressed until approximately 1989–1990, which did not provide a complete picture of how much contamination was deposited on the territory of the former Soviet Union. Only now, as new release data and analyses of fuel masses remaining within the sarcophagus are being made available, is it clear that earlier western suspicions were justified.
 20. U.S.S.R. State Committee on the Utilization of Atomic Energy, Volume II: Accompanying Material, Appendix 4, Appraisal of the Quantity, Composition and Release Dynamics of Radioactive Materials from the Damaged Reactor, Table 4-13, p. 20. The Soviet claim that the data in Table 4-13 of this report have a +50% margin of error is doubted by some in the West. By implication, this calls into question the shape of the curve and hypotheses that attempt to explain the nature and dynamics of releases during the Active Phase of the accident as well as the reliability of the Soviet data presented in Vienna. These problems will be addressed in more detail in an upcoming article.
 21. Although lacking the new information about the fact that this material had not entered the core shaft, Powers et al. provide an excellent source-term analysis of the data presented by the Soviets in Vienna.¹ In contrast, a rather complex but poorly substantiated theoretical analysis of the supposed filtration cooling of the core through the dumping of the materials is presented in V. P. Maslov, V. P. Myasnikov, and V. G. Danilov, *Filtering Cooling Model for the Pile-Up in the ChAPP Accident Block, Fission Product Transport Processes in Reactor Accidents, Proceedings to the International Center for Heat and Mass Processes, Dubrovnik, Yugoslavia, May 22–26, 1989*. The conclusion section of this paper is particularly poor because it appears to conclude what the authors wanted in the first place: (1) that their filtration cooling model applies well to explain the drop and subsequent rise in total releases (hence giving the peculiar "bathtub" curve shape), (2) that "the filtration cooling model of the pile-up completely explains all the phenomena taking place in the reactor after the accident" (emphasis added), and (3) that "the convective cooling model convincingly explains, why [corium] fragments did not burn through the bottom of the reactor pit [Lower Biological Shield (LBS)]," when, in fact, the LBS was completely penetrated by the corium.
 22. Yu. V. Sivintseva and V. A. Kachalova (general editors), *Chernobyl: Five Difficult Years*, p. 251, Moscow, Publishing House, 1992. Evacuation of the more than 49 000 residents of Prypjat began 36.5 hours after the accident, and evacuation of the remaining towns and villages (including the town of Chernobyl with 12 500 residents) took more than a week. Altogether, from April 27 through August 16, 90 784 people were evacuated from 75 population centers in Ukraine—89 489 from the Kyjiv oblast and 935 from the Zhytomyr oblast.
 23. International Advisory Committee IAEA, *International Chernobyl Project: Technical Report, Evaluation of the Radiological Consequences and the Protective Measures*, p. 84, International Atomic Energy Agency, Vienna, 1992.
 24. Immediately after the accident, two young senior reactor control engineers-in-training, Aleksandr Kudryavtsev and Viktor Proskuryakov, were told to go to the Central Hall above the reactor and manually insert the jammed scram rods into the reactor. Upon arriving they saw that "A red and blue fire was burning from the mouth of the reactor, with a powerful updraft" [emphasis added, Gregori Medvedev, *The Truth About Chernobyl*, p. 103, Basic Books, New York, 1991]. This appears to confirm that the graphite was burning within the reactor shaft while supplied with oxygen by a powerful draft of air. That night, Shcherbina noted the same from a helicopter, "[he] looked through binoculars at the reactor, now bright yellow from the extreme heat. Against the background he could clearly see the dark smoke and tongues of flame. And a glimmering blue light, not unlike starlight, shone from deep gashes to both right and left, deep within the bowels of the destroyed core. It seemed as if some immensely powerful hand was pumping invisible bellows, fanning that gigantic nuclear furnace, 20 m in diameter" (p. 177).

25. From a private communication with Serhij Vasyli'ovich Shirokov, March 29, 1993, in Kyjiv.
26. Viktor Haynes and Marko Bojcun, *The Chernobyl Disaster*, p. 142, The Hogarth Press, London, 1988.
27. At some time during the bombing of Unit 4 with lead, sharp criticism was leveled against proponents of this idea. Melted and vaporizing lead, it was argued, would result in an ecological disaster through lead poisoning of the surrounding environment. Nonetheless, the bombing continued unabated until late on May 2. (Personal conversation with Aleksandr Aleksandrovich Borovoi, Head of the Division of Radiation and Nuclear Safety of the Inter-Agency Scientific and Technical Center "Shelter," May 1992.) This is also confirmed in Ref. 11, p. 133.
28. A. A. Borovoi, *Post-Accident Management of Destroyed Fuel From Chernobyl: Technologies Used and Lessons Learned, Chernobyl, Ukraine and Moscow, Russia*, p. 15, International Atomic Energy Agency, 1990.
29. For descriptions of bore sample collection and general research within the Sarcophagus, see S. T. Belyayev, A. A. Borovoi, Zh. G. Volkov, A. Yu. Gagarinskij, N. Ye. Kukharov, G. A. Sharovaro, and B. F. Shkalov (Eds.), *Technical Basis for Nuclear Safety of the Object "Shelter, Chernobyl, Ukraine"*, Complex Expedition of the I.V. Kurchatov Institute of Atomic Energy, 1990.
30. *Technical Basis for Radiation Safety of the Object "Shelter," Chernobyl, Ukraine*; Minsk, Belarus: Complex Expedition of the I.V. Kurchatov Institute of Atomic Energy, Institute of Radioecological Problems at the Belarusian Academy of Sciences, 1992.
31. Alexander R. Sich, *The Chernobyl Accident Revisited: Source Term Analysis and Reconstruction of Events During the Active Phase*, Chapter V.3, Nuclear Engineering Ph.D. dissertation, Massachusetts Institute of Technology, January 7, 1994.
32. *RBMK-1000 Reactor: Technical Description of the Construction*, a reference book for training RBMK operators; and Gostroj-State Committee on Construction in the USSR, *VNIET Blueprints*, Moscow, November 1976 and October 1979.
33. Personal conversations (September 24–28, 1992) with Konstantine Pavlovich Checherov, *Complex Expedition* scientist, and Vadym Vasylyovych Hryshenko, Head Construction Engineer for Units 5 and 6 called up to manage accident management actions. Actually, the dispatcher acted more as an air traffic controller for the dozens of helicopters engaged in the operation. There were no direct commands given to drop the material—only the crew members could see what they were targeting. The general command seems to have been to hit the "red glowing mass." Hindsight notwithstanding, one wonders why, instead of having commands radioed from a man 3 km away (who obviously could not see whether the bags of materials were hitting the intended target), another helicopter hovering at 1500 to 2000 m above the destroyed reactor could not have been safely used to issue the same commands, and it would have proved much more effective.
34. A. A. Borovoi, *Inside and Outside of the Sarcophagus*, p. 6, Preprint, Chernobyl, Ukraine: Complex Expedition of the I. V. Kurchatov Institute of Atomic Energy, 1990.
35. Spartak T. Belyayev, Alexand[e]r A. Borovoy[i], and I. P. Bouzouloukov, *Technical Management on the Chernobyl Site: Status and Future of the "Sarcophagus," in Nuclear Accidents and The Future of Energy: Lessons Learned from Chernobyl: Proceedings of the International Conference in Paris, France, April 15–17, 1991*, European Nuclear Society (ENS), Paris, France: European Nuclear Society, 1991, p. 27. "Between 1986 and 1989, unsuccessful attempts were made to find traces of lead in rooms below the reactor, the washing tank [pressure suppression pool], the steam distribution duct [corridor], the room under the reactor [sub-reactor region], etc. Only three years after the accident (1989), after many search attempts, was it realized that the problem posed could not have been solved, or at least only partially Under the most optimistic assumptions, only a small part of materials could really have been dumped in the reactor pit. Most formed mounds up to 15 meters high in the reactor control room [Central Hall]. Similarly, it was impossible to block all channels taken by air to escape from the reactor pit, in other words create a suitable filtering layer."
36. 1991 NOVA documentary (originally Horizon/BBC) entitled *Suicide Mission to Chernobyl*. Interestingly, without realizing its implications, the same claim is also made: "... the military attempted to bomb the reactor with neutron-absorbers and other chemicals. Several pilots flew straight through an invisible plume of radioactive particles and died soon after. Despite their valiant efforts, almost no neutron absorbers got into the core." [emphasis added]
37. Zhores Medvedev, *The Legacy of Chernobyl*, p. 60, W. W. Norton and Company, New York, 1990.
38. S. N. Begichev, A. A. Borovoi, E. V. Burlakov, A. Ju. Gagrinsky, V. F. Demin, I. L. Khodakovsky, and A. A. Khrulev, *Radioactive Releases Due to the Chernobyl Accident*, Presented at the *International Seminar on Fission Product Transport Processes in Reactor Accidents*, May 22–26, 1989, p. 9, Dubrovnik, Yugoslavia, 1989.
39. U. S. Nuclear Regulatory Commission (comp.), *Report on the Accident at the Chernobyl Nuclear Power Station*, Report NUREG-1250, pp. 6-3–6-8, 1987.
40. Private conversation with Vadym Vasylyovych Hryshenko, October 13, 1992. Because the Governmental Commission files are still under lock and key, this scenario can by no means claim to be definitive, although it is presumably the most accurate to date.
41. Chernobyl—The IAEA Visit, *Nuclear Engineering International*, June 1986, p. 3. Morris Rosen (at the time of the accident Deputy Director of the Division of Nuclear Safety in the IAEA's Department of Nuclear Energy and Safety), based on Soviet information, stated that "other measures being taken to bring the situation under control include the use of gaseous nitrogen from containers of liquid nitrogen introduced to the access area, to maintain an inert atmosphere in the reactor area and to assist in cooling." It is not clear what was meant by "access area" nor how gaseous nitrogen could provide effective cooling or displacement of oxygen in a reactor building the size of Chernobyl Unit 4, which has dimensions approximately $70 \times 70 \times 70 \text{ m}^3$ on a side. There is actually much information in this article (which the Soviets provided) that is untrue or could not have been known so early after the accident given the extreme conditions in the area of Unit 4.
42. A. A. Borovoi, G. D. Ibrahimov, S. S. Ogrodnik, V. D. Popov, and K. P. Checherov, *The Status of Chernobyl Unit-4 and its Nuclear Fuel (As a Result of 1988–1989 Research)*, Preprint, Chernobyl, Ukraine: Complex Expedition of the I.V. Kurchatov Institute of Atomic Energy, 1990.
43. Michael J. Driscoll and Frank L. Bowman, *Core-Catcher for Nuclear Reactor Core Meltdown Containment*, U.S. Patent 4113560, September 12, 1978.

44. C. W. Forsberg, E. C. Beahm, and G. M. Parker, Core Melt Source Reduction System (COMSORS) to Terminate LWR Core-Melt Accidents, in *Proceedings of the 2nd International Conference on Nuclear Energy (ICONE-2)*, San Francisco, Calif., March 21–24, 1993.
45. Ronald Allen Knief, *Nuclear Engineering: Theory and Technology of Commercial Nuclear Power*, 2nd ed., p. 403, Hemisphere Publishing Group, Washington, D.C., 1992.
46. James Varley, Who Was to Blame for Chernobyl?—INSAG's Second Thoughts, *Nuclear Engineering International* (May 1993), p. 51. The Soviet delegation's suppression of information vital to understanding the causes and consequences of the accident, "... now looks more like a matter of deliberate deception."

General Safety Considerations

Edited by G. T. Mays

The IAEA-ASSET Approach to Avoiding Accidents is to Recognize the Precursors to Prevent Incidents

By F. Reisch^a

Abstract: *The International Atomic Energy Agency (IAEA) runs the Assessment of Safety Significant Event Team (ASSET) services for assessing safety performance at nuclear power plants. The aim of the ASSET missions is to prevent incidents and accidents. The way to achieve this is through the analyses of the safety relevant events experienced during operation, testing, and maintenance.*

All major accidents had precursors; this has been proved for the 1978 Three Mile Island¹ and the 1986 Chernobyl² accidents. An effective method to avert accidents and avoid repetition of unpleasant events is to appreciate and fully use the lessons that can be learned from the safety relevant events that occurred during operation. These events must be analyzed in depth by plant personnel while preparing to present them to the dozen or so International Atomic Energy Agency (IAEA) Assessment of Safety Significant Event Team (IAEA-ASSET) mission members coming from all corners of the world.

An ASSET team consists of about a dozen professionals, each with extensive experience in the operation and design of nuclear power plants. Usually two of them are from the IAEA. The team members are all experienced operators and regulators, often in managerial positions, who are well aware of the problems plant personnel are facing. They are always engineers, never psychologists.

An IAEA-ASSET mission is unique because in a 2-week period several hundred safety relevant events are reviewed, three that are typical are reviewed very extensively, conclusions concerning pending safety issues are drawn, and a comprehensive report is issued.^{3,4} The most important results obtained by the international experts are concrete suggestions aiming for improvements that can be accomplished in a short time for a low cost. ASSET does not accept excuses putting the blame on parties outside the plant but recommends basically only remedies that can be accomplished by the plant management. Design modifications might be necessary; however, software-related measures, such as improved procedures and operator training, are the only ones that can be implemented quickly with limited resources and therefore have particular interest for plant management.

ASSET provides other services also, such as training courses, preparatory missions, and follow-up missions. These services require less time and smaller teams. Up to now, 24 countries invited ASSET to provide various services.

THE ASSET METHODOLOGY

The ASSET methodology has been described in a report from a recent IAEA meeting.⁵ The skeleton information given here contains the terminology to facilitate understanding of the examples from the Rovno and Leningrad nuclear power plants.

^aSwedish Nuclear Power Inspectorate.

The ASSET event investigation answers three questions:

1. What happened?
2. Why did it happen?
3. Why was it not prevented?

The answer to the third question is sensitive; however, it is important to answer to prevent recurrence of the event.

An event is defined as something that happened unexpectedly during operation or maintenance. An event is made up of a series of occurrences that finally led to the undesired behavior of the plant. During the course of a mission, after screening several hundred operational and maintenance events, three are chosen to represent typical pending safety issues for in-depth analyses. The selection of these three events requires good insight, sensitivity, and perception of the experts. Awareness of the results of probabilistic safety analyses (PSA) is an advantage; however, often local conditions revealed during the course of the mission point to problems that demand different priorities. Crucial safety issues must be highlighted through the analyses of these events to ask for urgent remedies. Because of the limited time available, the experts form three subgroups, one for each event. For each selected event, a narrative is prepared, the chronological sequence of the event is established, and afterwards the logic tree of occurrences, which leads to the event, is prepared. A few of the occurrences are investigated in detail with the root-cause analyses method.⁶

A well-defined terminology is applied to make it easier for the team members and the plant personnel to have a common understanding of the meaning of the words they are using when they communicate to apply the root cause analyses method:

1. Occurrence refers to "What failed to perform as expected?"
2. Direct cause refers to "Why did it happen?"
3. Root cause refers to "Why was it not prevented?"

The types of failures are systematized according to the following three categories:

1. Equipment failure
2. Procedure failure
3. Human error

Because of IAEA priorities and the wish of the Eastern European countries to become familiar with western practices, most of the power plants scrutinized by ASSET are equipped with the Russian-designed reactor types, VVER (pressurized-water reactor) and RBMK (graphite-moderated boiling-water-cooled pressure tube reactor,

like the one in Chernobyl). The design aspects of these reactors have been analyzed in detail in *Nuclear Safety* and in other Western publications (e.g., see Refs. 7 to 9). However, their operational performance has not yet been presented in any extent for the Western public. It is appropriate now that the Western countries take advantage of the transparency provided by the VVER and RBMK operators through learning about the use of the ASSET methodology in Eastern Europe. Each ASSET report is at least a couple of centimeters thick; to keep this article short, only some of the highlights from two of the reports follow. The examples are one event from the Rovno nuclear power plant in Ukraine in the Rovno city region and one event from the Leningrad nuclear power plant¹⁰ in Russia in the St. Petersburg city region. Also, the application of the root-cause analysis is briefly demonstrated, and extracts of the Recommendations of the ASSET teams at these two missions are given. The complete reports, like the other ASSET reports, include also flow diagrams, electrical schemes, drawings, and design information and are available from the IAEA.

ROVNO ASSET MISSION TO THE TWO VVER-440/213 PLANTS IN DECEMBER 1993

At Unit 2 an interesting event happened on Dec. 13, 1992, namely, an "Excessive rate of decrease of temperature and overcooling of primary circuit following failure of valves, induced and enhanced by incorrect operator action." This was one of the three events chosen for further analyses and will be briefly presented here. The other two events analyzed dealt with electrical failures and human errors.

Description of the Event¹¹

When operating Unit 2 at nominal power, problems were encountered with the feedwater control to steam generator-4 (SG-4). In the process of correcting this problem, SG-5 inadvertently was switched to manual feedwater control. This remained unnoticed until low level at SG-5 was signaled. To raise the level, the feedwater supply was increased to full. Inadequate manual control subsequently resulted in the signals "high level in SG-5" and "too high level in SG-5." The protection system intervened by closing the shutoff valves to the turbines and initiating SCRAM. After the automatic opening of the turbine bypass valves and a steam dump valve to atmosphere, the steam pressure started to fluctuate, and an asymmetrical steam flow appeared. This

caused the protection system to be activated over $dp/dt > 0.09$ MPa/s. The protection system de-energized power supply to one vital bus bar and initiated the startup of the emergency power supply system from DG-3 and also initiated isolation of the power conversion steam system (turbine system). One emergency core cooling system/high-pressure safety injection (ECCS/HPSI) train was activated. One fast-acting pneumatic isolation valve failed to close as expected, and the opened steam dump valve failed to close on automatic because its instrumentation and control (I&C) system was temporarily de-energized. A safety valve of SG-3 opened and failed to close when steam pressure decreased. This went unnoticed at that time. The steam dump valve was closed on manual. The one activated ECCS/HPSI train was stopped. Late recognition of the actual plant situation and inadequate operator action enhanced by inadequate procedural support resulted in an excessive cooldown rate of over 30°C/h and finally a primary circuit temperature of 208°C . After more than 45 min, the correct action of isolation of SG-3 was performed, and the unit was stabilized, normalized, and brought to the cold shutdown state. Possible affected areas were inspected.

To win precision and accuracy, the next step in the ASSET process is the establishment of the chronological sequence of the event (Table 1).

When the experts reach a clear understanding of the event, they define the individual occurrences and assign to each of them the type of failure that is dominant. The series of failures leads to the event. Figure 1 shows the form used by the experts to establish the logic tree. The * sign indicates the occurrences that were chosen by the subgroup for detailed root-cause analyses.

To display the tools the experts are using, Figs. 2 and 3 give the forms that were filled in for a procedure failure, "Operating instructions failed to give adequate guidance to address secondary steam leaks" (occurrence 6), and for a personnel failure, "Operator failed to isolate SG-3 timely" (occurrence 7).

After screening all the events, the experts, on the basis of their collective knowledge, summarized their recommendations under these headings:

1. Recommendations to optimize the balance between software and hardware safety provisions.

- a. Plant management should review the content of the periodic training program for operations staff, particularly those staff who may be required to carry out tasks infrequently, such as the shift supervisor required to carry out switching operations during a rapid plant transient.

Table 1 Sequence of Events on Dec. 13, 1992, Rovno 2

Time	Event
18.39.29	Nominal power. Feedwater controller SG-4 observed to be defective. Incorrect engaging of bypass controller and inadvertent switching off SG-5 feedwater control to manual.
18.39.37	SG-5 level "too high," protection system activated turbine trip and SCRAM.
18.42.42	Opening of turbine bypass valves.
18.42.44	Automatic opening on SG-3 for one of the two steam dump valves to atmosphere. Steam pressure fluctuations and asymmetric flow.
18.42.56	Actuation of the safety system on $dp/dt > 0.09$ MPa/s.
18.43.00	All pneumatic isolation valves closed except one.
18.43.12	The previously de-energized bus bar is re-energized from DG-3.
18.43.37	Safety valve on SG-3 opened. Steam dump valve to atmosphere failed to close automatically.
18.44.00	Operators close steam dump valve on manual.
18.45.34	All electric-driven isolation valves closed.
18.47.02	ECCS/HPSI-3 stopped.
19.00	Safety valve on SG-3 observed to be not closed.
19.42.51	Isolation valves closed in the primary circuit loop of SG-3.

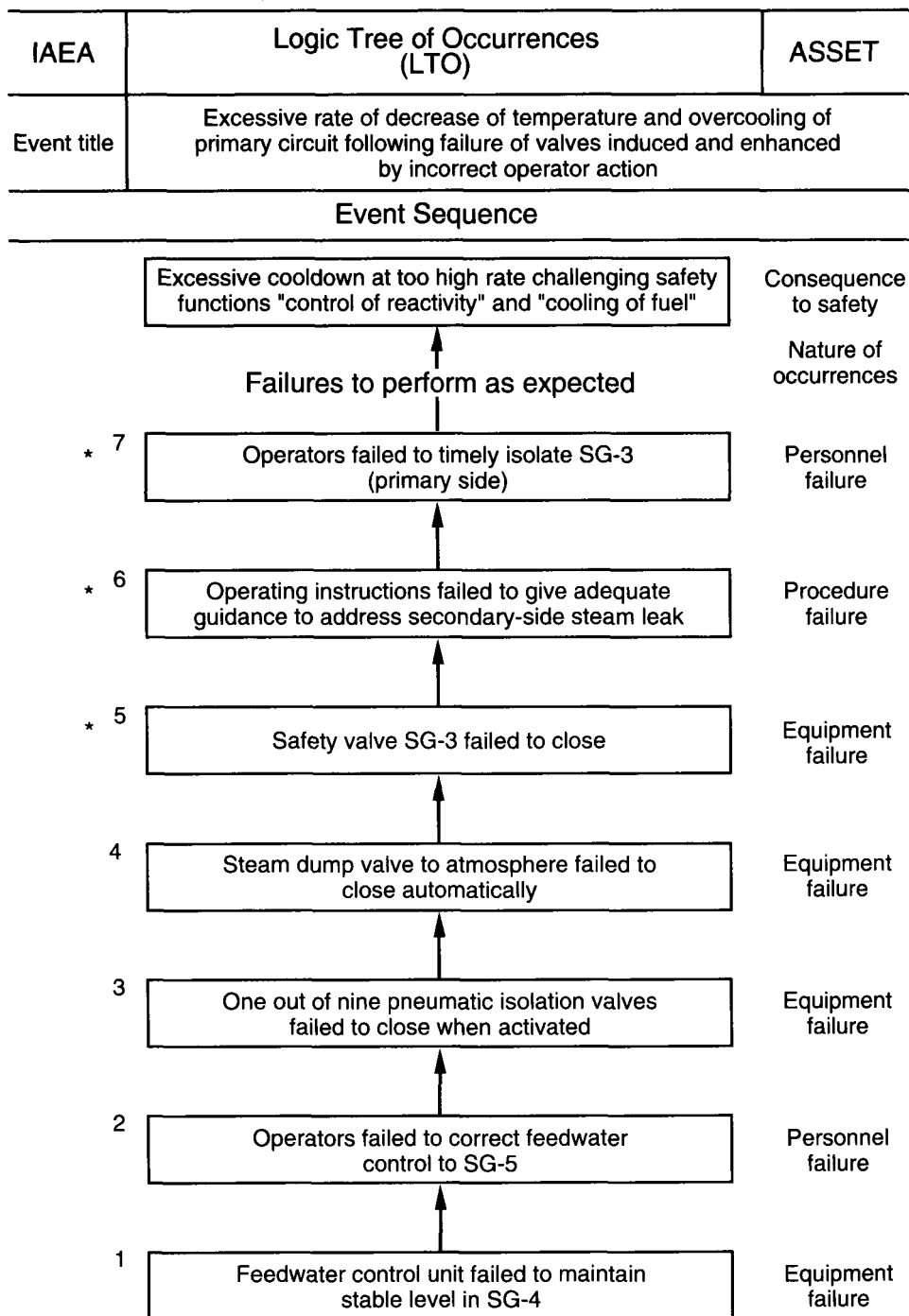
- b. Plant management should enhance the comprehensiveness of maintenance procedures by including all the steps required. Where the plant has to be put into abnormal states or configurations to carry out maintenance work, operations staff should be involved in the preparation and validation of maintenance procedures.
- c. Plant management should ensure that operating instructions are amended promptly and should instigate a system for the issue of properly authorized plant temporary instructions where urgent amendments are required.
- d. Plant management should review the need for the high number of instances where reactor safety protection lines are disconnected for maintenance activities.

2. Recommendations to improve the plant programs for prevention of latent weaknesses.

3. Recommendations to improve the feedback from operating experiences.

To keep the size of this article reasonable, no details of the last two groups of recommendations are given.

The most examined reactor type by ASSET is the VVER-440 with 11 missions, and out of them, 4 followups by the end of 1993. These reactors



*Occurrences which were chosen by the subgroup for detailed root cause analyses.

Fig. 1 Logic tree of occurrences.

IAEA	Event Root Cause Analysis Form (ERCAF)	ASSET							
Event title	Excessive rate of decrease of temperature and overcooling of primary circuit following failure of valves induced and enhanced by incorrect operator action	December 13, 1992							
Occurrence: What failed to perform as expected?		Corrective actions by plant							
Occurrence title	Operating instructions failed to give adequate guidance to address secondary steam leaks (*6)								
Nature of the failure	Procedure failure	Appropriate		Comprehensive		Implemented			
Direct cause: Why did it happen?		Corrective actions by ASSET		Yes	No	Yes	No	Yes	No
Latent weakness of the element that failed to perform as expected	Inadequate instructions in post-scrum procedures to timely detect and to react to secondary-side steam leaks	I Amend post-scrum instructions with regard to detection of and reaction to secondary steam leaks		X		X			X
Contributor to the existence of the latent weakness	*Deficiency of quality control prior to operation? *Deficiency of preventative maintenance? Deficiency in quality control in process of production of operating instructions	II Implement quality control for technical contents in process of production of operating instructions		X		X		X	
Root cause: Why was it not prevented?		Corrective actions by ASSET							
Deficiency to timely eliminate the latent weakness	*Inadequate detection, analysis, repair, remedy? Inadequate repair of post-scrum instructions based on operating experience feedback due to hesitance to issue essential instructions	III Establish promulgation of provisional operating instructions		X		X			X
Contributor to the existence of the deficiency	Inadequate management policy for: • Surveillance • Operating experience feedback? Standard operating practice does not sufficiently emphasize using authority provisionally issue operating instructions	IV Encourage and emphasize "top down" to exercise the existing authority to the fullest		X		X			X

Fig. 2 Event root-cause analyses form (occurrence 6).

never experienced an accident like TMI-2. The absence of cracks in the primary circuit was also noted, and all the original steam generators are still in service, some of them for more than two decades. Numerous equipment and procedure failures and also human errors occurred

with this reactor type without causing any accident or fuel failure. Considering the many reactor pressure-vessel head cracks and steam generator replacements at Western PWRs, the Russian designers and operators feel that they have accomplished a remarkable achievement.

IAEA	Event Root Cause Analysis Form (ERCAF)	ASSET							
Event title	Excessive rate of decrease of temperature and overcooling of primary circuit following failure of valves induced and enhanced by incorrect operator action	December 13, 1992							
Occurrence: What failed to perform as expected?		Corrective actions by plant							
Occurrence title	Operator failed to isolate SG-3 timely (*7)								
Nature of the failure	Personnel failure	Appropriate	Comprehensive	Implemented					
Direct cause: Why did it happen?		Corrective actions by ASSET		Yes	No	Yes	No	Yes	No
Latent weakness of the element that failed to perform as expected	Inadequate proficiency of operators to adequately and timely react to secondary side anomalies	I Specific instruction of all operating personnel and additional "on the job" training		X		X		X	
Contributor to the existence of the latent weakness	*Deficiency of quality control prior to operation? *Deficiency of preventative maintenance? Inadequate proficiency check	II Proficiency check upgraded		X		X		X	
	Inadequate training	Simulator training syllabus upgraded (Greifwald)		X		X		X	
Root cause: Why was it not prevented?		Corrective actions by ASSET							
Deficiency to timely eliminate the latent weakness	*Inadequate detection, analysis, repair, remedy? Inadequate detection of deficiency in proficiency check	III Review of proficiency check periodically and based on feedback of experience Review of training program periodically and based on feedback of experience		X		X		X	
Contributor to the existence of the deficiency	Inadequate management policy for: • Surveillance • Operating experience feedback? Deficiency in management policy for surveillance of proficiency checks and operator training	IV Management attention Enhanced policy amended		X		X		X	

Fig. 3 Event root-cause analyses form (occurrence 7).

LENINGRAD ASSET MISSION TO THE FOUR RBMKs IN MAY 1993

At Unit 3 on Mar. 24, 1992, an event happened that echoed across the whole world, namely, a "Fuel damage followed by release of unfiltered gases outside the plant." This was one of the events analyzed by ASSET.

Description of the Event¹²

On Mar. 24, 1992, Unit 3 was operating at full power. At 2:34:40 the flow of water to one channel decreased sharply from above 20 m³/h to below 3 m³/h. The channel tube ruptured and in about 5 seconds high core cavity pressure initiated a fast emergency reactor trip,

a turbine generator shutdown, and closing of the flow of helium and nitrogen to the core cavity. In addition, the fuel channel integrity monitoring system showed high moisture levels in the core cavity.

Seven seconds after low water flow, response of the three main system areas was as follows:

1. *Reactor cooling system:* After about 3 minutes, the flow through each of the main circulation pumps was reduced by the operator to about 6000 m³/h by adjusting the main discharge valves. Also, the diesel generators were started up automatically and the system operated correctly throughout the event, whereas the water losses (about 55 tons in the first hour and less afterwards) were compensated for.

2. *Water disposal system:* The boiling water from the bottom of the core cavity flowed to the active water collecting vessel. Because of the pressure reduction, a gaseous mixture evaporated from the liquid. The mixture comprising steam and radioactive gases escaped untreated from the vessel because of an open vent. This, in turn, allowed active gases to be detected by the radioactivity monitoring system 2 minutes after the reactor trip. As a consequence, the radiology department issued protection masks, respirators, and potassium iodide tablets to the personnel present.

3. *Steam disposal system:* The large quantity of steam, with radioactive gases, from the core cavity flowed through the first water seal in about 10 seconds. This caused pressure buildup in the "steam path to the localization system" and thus resulted in flow through the second water seal (a large suppression pool) after about 30 seconds. Gradually, a slight pressure buildup occurred in the large localization system, and a start of emission to atmosphere occurred via the vent pipe on top of the localization building. In this emission, radioactivity was measured 30 minutes after the channel rupture. Follow-up actions on this occurrence were:

- The incoming ventilation systems were switched off (50 minutes after channel rupture).
- Actions were started to reroute the gas to the filtration system. This involved operation of certain valves in the filtration, opening the valves between the localization system and the filtration system, followed by closing of the valves to the vent pipe. This was effected at 3:40 (65 minutes after the channel rupture).

For easy understanding, a simplified flow diagram (Fig. 4) indicating the major components mentioned in

the description is given. A complete picture of the event is far beyond the available space in this publication. For a flavor of how the actual ASSET subgroup was working, Fig. 5 shows the logic tree of occurrences in the form created by the group members.

A serious deficiency was the lack of adequate procedures. Therefore the one form out of several root-cause analyses shown here (Fig. 6) deals with the occurrence "Procedures fail to give guidance for manual switch over to filtered venting."

The members of the ASSET team were well aware of the fact that the plant management and personnel inherited a design that left them much to do. The recommendations contain several pages, so it is not possible to quote them here. As a recognition of the great efforts made and the accomplishments already achieved, however, the preamble of the recommendations states that ASSET was mindful of the fact that there has been a continuous effort at the station to improve safety over the years since Chernobyl.¹³ Major reconstruction is in process, and more is to come. ASSET gave some examples: establishment of a new safety department, human factors and personal training departments responsible also for cross departmental investigation of events, improvements in the core kinetics (reduced void coefficient), improved time response of the fast-acting safety rods, and procurement of a site-specific simulator.

Some ASSET remarks about RBMKs led to immediate results, such as Brookhaven National Laboratory and the Moscow-based Design Institute are joining forces to improve the construction of the end caps of the fuel channels to avoid the primary steam leaking into the reactor hall often observed by ASSET at these reactors. Everything is not noted in the ASSET reports however (e.g., oscillations have been observed by operators). Several types were observed: one with a period of a few seconds similar to the coupled neutronics-hydrodynamics ones monitored in boiling-water reactors (BWRs) and another with a period of some 10 minutes as the result of temperature fluctuations observed in large graphite-moderated gas-cooled reactors.¹⁴ As in Western BWRs, to reinforce safety, it would be advantageous to introduce stability monitors in the control rooms. Transients with a trip of one of the two turbines are usual. Also, there are other reasons, such as shortage of fuel supply, which result in operation with partial power. It is known that these reactors are less stable at partial power than with full power, a fact that makes stability monitoring desirable.

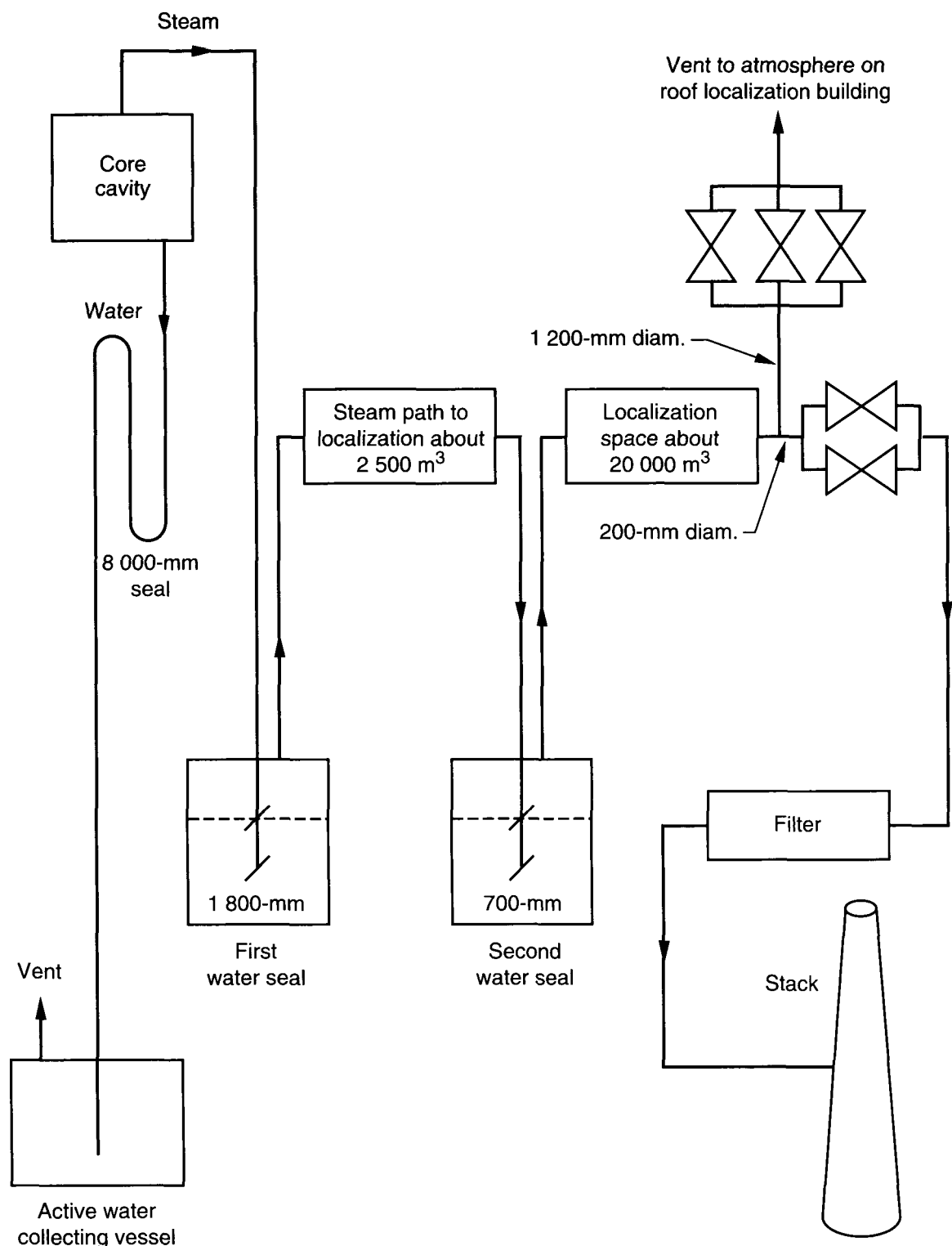


Fig. 4 Simplified flow diagram.

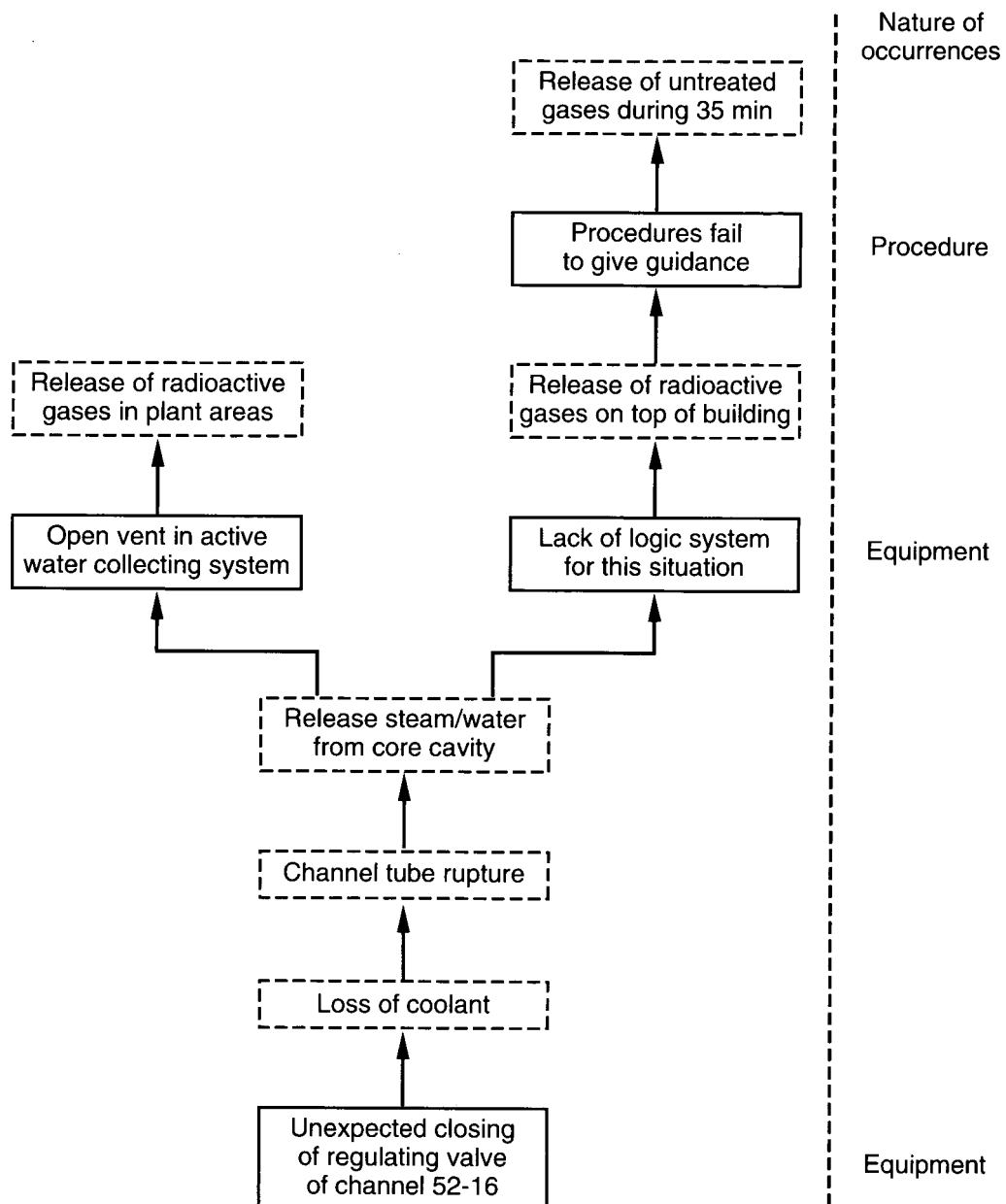


Fig. 5 Logic tree of occurrences.

IAEA	Event Root Cause Analysis Form (ERCAF)		ASSET
Event title	Fuel damage followed by release of unfiltered gases outside the plant		
Occurrence selected	(Failure to perform as expected) Procedures fail to give guidance for manual switchover to filtered vent		
Nature of the occurrence	(Equipment, personnel, procedure) Procedure deficiency		
Direct cause		Corrective action	
Latent weakness	Technological Procedures (TPs) were not written to cope with failure or nonoperation of localization system, thus resulting in delay of air release to filtered vent		To review TPs for content with respect to design basis requirements
Contributor to the existence of the latent weakness	The complex shift organization and divided responsibilities made it difficult to recognize and act on the problem		To align shift organization and responsibilities with the need for fast response to accidents
Root cause		Corrective action	
Deficiency to timely eliminate the weakness	Operations staff were not aware that localization system logic would not operate for a channel rupture		Reinforce training for operating staff in safety system functions related to design basis requirements
Contributor to the existence of the deficiency	Lack of clear design basis document to clarify the systems operation, specifically for channel rupture		The design basis for all the safety and safety-related systems should be available to plant staff

Fig. 6 Event root cause analyses form.

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Accident Analysis

Edited by R. P. Taleyarkhan

A Review of the Available Information on the Triggering Stage of a Steam Explosion

By D. F. Fletcher^a

Abstract: *This article reviews the available experimental data and modeling work on the triggering stage of a steam explosion. The importance of the triggering stage and the various different means of triggering an explosion are discussed. The extant modeling work is then reviewed, and it is concluded that no validated model exists. Data are reviewed from experiments designed to investigate triggering and the triggering behavior observed in medium-scale experiments using prototypic melts is examined. An attempt is then made to draw these data together and to come to some conclusions on the likely use of triggering arguments in steam explosion assessments. As an example, the use made of triggering arguments in the Sizewell B steam explosion assessment is presented.*

The main conclusions are that the data available do not support the hypotheses that early triggering is virtually certain and that triggering at high pressure is impossible. The available data do, however, suggest that triggering becomes more difficult with increased pressure and easier with increased water subcooling. Experimental results show that even a small increase in pressure from 0.1 to 0.5 MPa can inhibit spontaneous triggering.

It is concluded that the available data and modeling do not justify making strong claims using triggering arguments in steam explosion assessments. They do, however, allow modest claims for the lack of an effective trigger and allow the relative likelihood of triggering during the various stages of the melt-water interaction to be estimated.

The purpose of this article is to compile the available information on the triggering stage of a steam explosion

and to make judgments on the use of triggering arguments in probabilistic safety assessments. The role of steam explosions in nuclear reactor safety has been reviewed by, among others, Cronenberg¹ and Corradini et al.² In the past, most work was performed to address the α -mode failure issue, a postulated event in which an in-vessel steam explosion causes the reactor vessel to fail and generates a missile that causes the containment to fail, which leads to a release of fission products to the atmosphere. Steam explosions are also relevant in other situations, however. If a melt-through of the reactor vessel occurs, the melt could contact water in the sump of a pressurized-water reactor (PWR)³ or in the suppression pool of a boiling-water reactor (BWR),⁴ and a subsequent steam explosion could cause further damage to the plant. In addition, accident management procedures in both reactor types consider the likelihood that the chosen action will trigger a steam explosion.^{5,6}

A conventional steam explosion is generally considered to involve a progression through the stages of coarse mixing, triggering, propagation, and expansion.^{1,2,7} Triggering is the event that initiates the rapid, local heat transfer and pressure rise that is necessary if a propagating wave is to develop and lead to the rapid transfer of heat from the melt to the water. Experimental observations suggest that triggering is associated with the local collapse of the vapor layer around a melt droplet followed by rapid fragmentation of the droplet.^{2,8} Vapor film collapse may occur for a number of reasons.

First, vapor film collapse may occur if the interface contact temperature between the melt and the water falls

^aCurrent address: Department of Mechanical and Mechatronic Engineering, University of Sydney, NSW 2006, Australia.

below the minimum film boiling temperature, at which point the vapor film becomes unstable. This mode of collapse often occurs in experiments in which simulant materials (e.g., tin–water droplet experiments) are used.⁹ The collapse can occur either as the melt surface temperature falls as the result of heat transfer to the water or as a melt droplet moves into a zone in which the water temperature is lower. It is unlikely to be an important mechanism in core melt–water interactions because of the high melt temperature. Coarse-mixing simulations show that the melt will not cool sufficiently for vapor film collapse to occur during a fall through several meters of water in which melt is distributed in the form of droplets within a mixture of steam and water.¹⁰ In this situation, when film boiling does destabilize as the result of cooling of the melt, the droplet surface temperature would be significantly below its freezing point, and triggering would be unlikely to occur. Thus, if the melt superheat is low, a frozen shell may form around the outside of the droplets and prevent fragmentation when the melt contacts the vessel base.⁸

Second, vapor film collapse may occur because water is forced into contact with the melt. This may be caused by an applied pressure pulse (the usual experimental means of triggering an explosion), forced flow of water collapsing the vapor film, or local coolant entrapment. In the first case, the pressure pulse induces a particle velocity in the coolant, toward the melt, at the liquid–vapor interface. If this motion is sufficient to drive the water into contact with the melt, triggering occurs. In the second case, the bulk flow of water past a droplet (without a pressure wave being present) causes the vapor layer to be convected away from the melt and thus causes film collapse. This mechanism is likely to be important in situations in which the vapor film is thin because of the melt temperature being low or the water being highly subcooled. In the third case, water is entrapped within the melt or against the vessel wall by the melt and is superheated until its temperature rises to the homogeneous nucleation temperature, when it flashes into steam and thus throws the melt surrounding it into contact with water, which causes triggering. Examples of experiments in which these various mechanisms have been observed are discussed later in the article.

Explosions that result from a known trigger are usually referred to as triggered explosions, and those occurring because of some uncontrolled event are usually referred to as spontaneous explosions. If the trigger is provided by some artificial means, such as a detonator, the explosion is said to be externally triggered. Schins¹¹ has described some of the means (e.g., exploding

bridgewire and release of compressed gas) that have been used to trigger an explosion in experiments.

It is clear from the previous description that triggering is a complex phenomenon that is difficult to model and quantify. There is also clearly an element of randomness in the occurrence of spontaneous triggering. The purpose of this article is not to try to understand the detailed physics of triggering but rather to determine what useful data on triggering can be extracted from the available literature for use in steam explosion assessments. This study was motivated by the need to produce a quantification of the probability of α -mode failure for the Sizewell B PWR at the full range of possible pressures.¹² An attempt is made to answer the following questions:

1. Is early triggering as likely at low pressure as some workers claim?
2. Is there any reliable evidence on the effect of pressure on triggering?
3. Is it possible to draw any general conclusions on the factors that affect triggering in any given situation?

The remainder of this article is devoted to the tasks previously described. The following sections contain (1) a review of the available modeling; (2) a discussion of the data from triggering experiments, including a discussion of the means to prevent triggering; (3) a description of various relevant integral steam explosion experiments and a summary of the results obtained; (4) a discussion of the implications of the data on the use of triggering arguments in steam explosion assessments together with some calculations of the magnitude of possible trigger sources; and (5) some conclusions and recommendations.

THEORETICAL WORK ON TRIGGERING

Most modeling attempts have followed a similar approach: a one-dimensional model consisting of a melt layer, a vapor layer, and a liquid slug is usually assumed (Fig. 1). The idealized geometry represents a section of the vapor film surrounding a melt droplet. Steady-state film boiling is assumed to be established before the arrival of a pressure wave at the liquid–vapor interface.

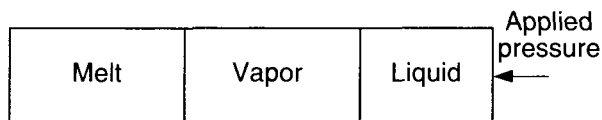


Fig. 1 Geometry used in the triggering modeling.

Conservation equations for mass, momentum, and energy are then used to model the transient evolution of the system. The main difference between the various models is in the level of complexity of the equation system used and in the physical processes modeled.

The earliest model appears to be that of Drumheller,¹³ who considered the symmetric collapse of film boiling around a sphere. The liquid was assumed to be incompressible, and energy considerations were used to derive an equation similar to Rayleigh's classical bubble collapse equation but with phase change terms. The assumption of a spherically symmetric collapse would appear to be questionable because of the finite time required for a pressure pulse to pass the sphere. No comparisons with experimental data were made using this model.

The first detailed model to be developed and compared with an experiment was that of Inoue, Ganguli, and Bankoff.¹⁴ In their model, a full nonequilibrium kinetic theory treatment of evaporation and condensation was used at the vapor-liquid interface and a Knudsen layer was modeled at the melt-vapor interface. The heat conduction equation was solved in the moving liquid slug by assuming a temperature profile that was a quadratic function of the distance from the vapor-liquid interface. A Newtonian model of the slug dynamics was used to determine the motion of the liquid. No vapor flow out of the film was modeled. The results of the calculations for liquid Freon 113 highlighted the importance of choosing the evaporation-condensation accommodation coefficient correctly. The paper contains a discussion of the effect of the presence of a permanent gas in reducing the accommodation coefficient because of the increased interfacial mass transfer resistance. (A recent paper by Barrett and Clement¹⁵ contains a detailed discussion of kinetic evaporation and condensation rates.)

A simplified model was also developed by Inoue, Ganguli, and Bankoff¹⁴ in which the heat storage in the vapor film was neglected and a heat balance was applied at the liquid-vapor interface to determine the condensation or evaporation rate, with the liquid-vapor interface temperature set to the local saturation temperature. This resulted in a much simpler set of model equations. The authors concluded that the full model was more reliable because it gave better agreement with data from Freon 113 vapor film collapse experiments. For any applied pressure pulse, the calculated response of the system was either total collapse of the film (judged to have occurred when the film thickness was of the same size as the surface roughness) or oscillation of the film thickness, with the pressure in the film rising sufficiently as collapse started to occur that it pushed the slug away.

This work was extended by Inoue et al.¹⁶ to allow for mass flow out of the film. The form of the mass flow term appears similar to that obtained in a standard film boiling model.¹⁷ The study showed that pressure pulses with steep fronts (i.e., shocks) were more effective at collapsing the vapor film than slow pressure rises. It was also noted that the collapse behavior was very sensitive to the ambient pressure because at higher ambient pressures more mass and energy are in the vapor layer.

A similar analysis has been pursued by Corradini,¹⁸ who also examined the effect of different mass transfer assumptions and came to the same conclusions as Inoue et al.¹⁴ He concluded that the equilibrium model was valid for shock rise times greater than 100 μ s. This conclusion was based on a comparison of the computed peak heat fluxes with measured values for experiments performed by Inoue and Bankoff¹⁹ (described later). He noted that the neglect of the Rayleigh-Taylor instability at the interface in the modeling would tend to reduce the predicted heat fluxes.

Kim and Corradini²⁰ have investigated the behavior of the vapor layer during its growth phase. Their analysis was for a spherical droplet of melt surrounded by a thin vapor layer initially. No loss of vapor (except by condensation) was allowed. As the vapor layer grew, the pressure oscillations in the system were studied. The effects of various parameters, such as initial vapor film thickness, ambient pressure, coolant temperature, and the presence of permanent gas, were examined. It was concluded that all these interact in a complicated way but that, in general, the parameters affected stability in the obvious way (e.g., thicker films were more stable). Their model showed that a small pressure increase, from 0.1 to 0.2 MPa, caused perturbations to the vapor-liquid interface to grow more rapidly, but as the pressure was increased further, the perturbations grew more slowly.

Knowles²¹ developed a one-dimensional model based on assumptions similar to those of Inoue et al.¹⁴ and Corradini.¹⁸ His model, however, used a more rigorous treatment for the slug dynamics and heat transfer into the vapor layer. He solved mass and momentum equations in the liquid slug so that its compressibility was taken into account and the detailed behavior of the incident pressure pulse could be modeled. Also, he solved a finite-difference form of the conduction equation in the liquid layer and the melt rather than assuming given temperature profiles. Equations from kinetic theory were used to simulate evaporation-condensation processes, with the kinetic theory equations being modified to allow for the net velocity of the interface. The pressure and temperature dependence of thermophysical properties was included.

Observations from simulations of the triggered collapse of low-pressure films around low-temperature surfaces led him to suggest the following criterion for collapse:

$$\tau_p > \delta(2p_{\text{trig}}/\rho c) \quad (1)$$

where τ_p is the duration of the trigger pulse, δ is the initial vapor film thickness, p_{trig} is the trigger pressure, ρ is the slug fluid density, and c is the sound speed in the slug. This equation is derived from the equations that govern sound wave transmission at an interface; if the particle velocity of the liquid at the slug-vapor interface multiplied by the duration of the pressure pulse is greater than the thickness of the vapor film, collapse will occur. It applies when film collapse is essentially unresisted.

Knowles' simulations showed that at higher melt temperatures stability was maintained by evaporation from the advancing slug. He argued that the vapor flux from this front would mix up any permanent gas present and that this would not produce a mass transfer barrier. In his simulations, the main effect of the gas is to increase the thermal conductivity of the film.

Attempts to apply this model to conditions relevant to the High-Pressure Thermite Rig (HPTR) experiments (see next section) were unsuccessful. The model failed to converge when the film became very thin. After examining the convergence history, he concluded that the fluid in the region of the vapor-liquid interface was in a metastable state that could not be handled by the existing framework of the model.²²

DISCUSSION OF TRIGGERING EXPERIMENTS

In this section the results from experiments performed specifically to study vapor film collapse and/or triggering are summarized. A description of experiments performed to determine ways of avoiding triggering is given in the following sections.

Simulant Experiments

Early experiments of a qualitative nature are described by Naylor.²³ These consist mainly of experiments to study the stability of a Leidenfrost drop on a heated surface.

Inoue and Bankoff¹⁹ investigated the triggered collapse of film boiling of Freon 113 or ethanol on an electrically heated nickel tube using a pressure step.

The magnitude of the pressure rise varied between 0.1 and 0.5 MPa, and the rise time of the pulse was varied between 80 μ s and 344 ms. Vapor film collapse was observed to occur when the pressure step had a magnitude greater than three times the ambient pressure and a rise time of <150 ms.

Inoue et al.¹⁶ investigated the triggered collapse of film boiling for the system of an electrically heated platinum foil immersed in water. The trigger used was a pressure step with a magnitude between 0.1 and 1.5 MPa with a rise time between 0.1 and 7.5 ms. The occurrence of collapse (or otherwise) was hard to detect in the experiments, but it appeared that partial collapse was triggered by a 0.5 MPa step and was more extensive as the pressure rise time was reduced.

Naylor²³ studied untriggered and triggered film boiling collapse on the surface of a brass rod with a hemispherical end immersed in a pool of water. Metal temperatures up to 770 K and water subcoolings ranging from 0 to 80 K were considered. The experiments were performed at ambient pressure. Naylor observed that the film could be collapsed by either a pressure pulse (generated by a shock tube in his experiments) or by the bulk flow of liquid. Observations suggested that collapse occurred when the average film thickness was less than the sum of the surface roughness plus the amplitude of interfacial waves on the liquid-vapor interface. By using a steady-state film boiling model to predict the vapor film thickness, he was able to confirm the collapse criterion given in Eq. 1. Thus, at low pressure and for low-temperature surfaces, vapor film collapse appears unresisted.

Corradini²⁴ has analyzed the data from over 300 single-droplet experiments performed by Nelson at Sandia National Laboratories (SNL). The experiments involved the release of small droplets of various melts (stainless steel, metallic corium, and oxidic corium) into water. The triggering behavior of the system was studied for various melt compositions, water temperatures, and ambient pressures. Two different triggers were used.

The first was an exploding bridgewire that yielded a peak pressure of 1 MPa at a distance of 40 mm and a rise time of approximately 1 μ s. The second was a detonator that yielded a pressure of 10 MPa at a distance of 40 mm and a rise time of approximately 20 μ s. Corradini's analysis of the data from these experiments led him to draw the following conclusions:

- Certain melt compositions did not lead to explosions, and this could be explained by the presence of incondensable gases, particularly hydrogen in the case of metallic melts.

- Explosions were suppressed at higher water temperatures and high ambient pressures because of the increased vapor film stability.

- Increasing the trigger magnitude can result in an explosion for a case in which an explosion did not occur for a weaker trigger.

Studies of the triggered fragmentation of melt droplets are being performed by Frost and coworkers^{25,26} and Theofanous and coworkers.²⁷ The studies performed by Frost et al. have highlighted the role of the relative velocity between the melt droplet and the coolant in determining the fragmentation mechanism. At low flow rates, a bubble growth and collapse mechanism is observed; at high flow rates, a stripping method is dominant. The experiments being performed by Theofanous and coworkers are aimed at the study of fragmentation dynamics following a propagation wave and use long-duration shock loadings. These experiments have highlighted the importance of "micro-interaction," the process in which the tiny fragments mix with some of the water present. Although both of these studies concentrate on fragmentation, they do highlight that the likelihood of an explosive event being triggered following vapor film collapse is a function of the local conditions.

Experiments performed at SNL have identified a trigger threshold for efficient fragmentation of a single droplet of melt.^{28,29} A three-order of magnitude increase in the volume of the vapor bubble formed following the fragmentation of a single melt droplet was observed when the trigger strength, defined as the product of the trigger pressure and impulse, exceeded a critical value. Berman and Beck²⁸ have developed an empirical parameterization of trigger strength, on the basis of an analogy with explosive welding, and postulate that the trigger strength can be characterized by the product of the trigger pressure and impulse. Currently, not enough detailed data are available to check their hypothesis. The work represents a significant step forward, however, because it is the first systematic study of the effect of trigger strength.

The importance of using realistic triggers in experiments if they are to be relevant to the reactor safety application has recently been discussed by Henry.³⁰ He has developed a simple empirical criterion, on the basis of the mixing energy required to cause more melt droplets to participate, to decide when an explosion that occurs following triggering is a propagating event or when it is simply energy released from the metastable mixture because of the fragmentation caused by the trigger. Thus care has to be taken to distinguish between experiments that use artificial triggers and reveal something about the

fundamental physics of steam explosions and those which use realistic triggers and indicate something relevant to reactor safety.

Effect of Pressure

The effect of ambient pressure on triggering is one of the crucial issues to be addressed in this review. In this section the available data from small-scale experiments and reactor incidents are collected and assessed. The following section deals with the HPTR experiments that were performed at Winfrith specifically to investigate the effect of pressure on triggering.

Experiments performed at JRC Ispra,³¹ in which molten salt was dumped into water, were found to lead to spontaneous explosive interactions at ambient pressure but not to explode at pressures of 0.5, 1, 2, and 4 MPa. In later work,³² various triggers were used—the strongest being a minidetonator charged with black powder. In this case, steam explosions were not observed for pressures above 3 MPa. These data have been used to justify the notion that triggering does not occur at high pressure. This is misleading. The experimental results suggest that spontaneous interactions are inhibited by increased pressure and that as the pressure is increased a stronger trigger is required to initiate an explosion. The correct conclusion to draw is that increasing the ambient pressure decreases the likelihood of spontaneous triggering *in this system* and that a stronger trigger than the one used in the experiments is required to initiate an explosion at pressures above 3 MPa. This suppression of spontaneous explosions at pressures above 0.1 MPa is consistent with the stability analysis of vapor film growth of Kim and Corradini.²⁰

A reactivity transient in an experiment being performed in the Power Burst Facility (PBF) at EG&G Idaho caused a fuel pin that was being cooled by water to fail; molten fuel was ejected into the coolant, and an explosion occurred at an initial pressure of 6.4 MPa.³³ A number of experimental reactors (e.g., NRX, BORAX 1, SL1, and SPERT-1D) have been damaged or destroyed as a result of this type of interaction.¹ In addition, it is believed that the Chernobyl reactor accident was initiated when a reactivity incident fragmented fuel, which was then expelled into two-phase coolant at a pressure of about 7 MPa.³⁴ However, it is important to realize that explosions of this type, where a reactivity incident causes highly fragmented fuel to come into contact with coolant, are very different from a conventional steam explosion. The stages of mixing, triggering, and propagation are bypassed; one can learn nothing about triggering from these incidents.

In the Three Mile Island-Unit 2 (TMI-2) accident, approximately 10 tons of melt relocated into the lower head of the vessel, which was filled with water, when the pressure was at 10 MPa.³⁵ It is not clear what the water temperature was because the high-pressure injection system had operated before this event. There was no evidence of a steam explosion, but the pressure was recorded as rising to 12 MPa. The time resolution of this rise is very coarse because the instruments were not designed for this situation. Although an explosion did not occur in this situation, it is not clear whether a coarse mixture ever developed and, therefore, whether triggers occurred but did not propagate or whether triggering did not occur.

Experiments in the High-Pressure Thermite Rig

A series of experiments was performed in the HPTR at Winfrith to study the triggering of steam explosions from ambient pressure up to 15 MPa.³⁶ In the experiments, 5 kg of a thermite-generated uranium dioxide-molybdenum melt at an initial temperature of 3600 K was poured into a 1-L pool of water contained in a pressure vessel. No assessment of the state of mixing was possible in the experiments.

The experimental series consisted of a program of 14 experiments. Three different triggers were used: (1) a pressure pulse, (2) a small water flow (4 mL), and (3) a large water flow (300 mL). The pressure pulse trigger was produced with the use of a water-helium shock tube. There are no reliable estimates of the magnitude and duration of the shock pulse. In the flow triggering experiments, the shock tube was used to inject a slug of water. In the small flow case, this water was at a temperature close to that of the pool; in the large flow case, however, water at ambient temperature, followed by helium, was injected. The experimental conditions are given in Table 1.

The only trigger to have any effect was the large water flow. In experiment 05 it initiated a steam explosion, and in experiments 13 and 14 it led to an increased steaming rate. In experiment 05 an explosion was triggered when the pressure in the vessel was 5.8 MPa (the pressure rise before the explosion was caused by mixing of water and melt in the confined volume). The pressure transient in this interaction had a rise time of the order of 100 μ s and created debris with 13% of the melt particles (by weight) being less than 500 μ m. The maximum pressure was 8.6 MPa. Thus, although there is little doubt that an explosion was triggered, it did not propagate through a significant fraction of the mixture.

Table 1 Test Conditions for the HPTR Experiments

Experiment	Pressure, MPa	Subcooling, K	Trigger ^a
01	1.0	20	N
02	1.0	20	N
03	2.1	19	N
04	2.1	19	P
05	2.1	20	LF
06	5.0	19	SF
07	10.0	85	SF
08	15.2	95	SF
09	10.3	64	SF
10	2.1	19	SF
11	2.0	15	SF
12	2.1	14	SF
13	2.1	20	LF
14	2.1	17	LF

^aN, no trigger; P, pressure pulse; SF, small flow; and LF, large flow.

It is very hard to draw any firm conclusions from the data provided by these experiments. The fact that in most cases no explosion was triggered could simply be due to the state of premixing at the triggering time. In addition, only one experiment was performed with the use of a pressure pulse as a trigger. In the one explosion that was triggered, a slug of cold water and helium gas were injected into the interaction vessel. Because this form of triggering is unlikely to occur in a reactor accident, this experiment does not contradict the view that triggering is likely to be more difficult at high pressure.

Methods of Avoiding Triggering

In this section, a brief discussion of practical methods for avoiding explosions is given. The first work in this area was the classic study of aluminum explosions by Long.³⁷ He performed an extensive series of experiments in which molten aluminum was poured into a tank of water. He observed that if the melt was prefragmented (using a wire grill) before contact with the water, it was less likely to explode than if it was released as a coherent mass. He found that coating the base of the container with lime or gypsum or allowing it to rust led to explosions in situations that would not have led to an explosion if the vessel base had been made of uncoated, degreased steel. Conversely, explosions did not occur if the vessel base was coated with grease or oil. Explosions were also avoided by painting the vessel base with a bituminous paint (called Tarsel). These treatments obviously affect

the wettability of the surface and change the likelihood of entrapping water beneath a puddle of melt. This work has been developed further by Nelson et al.,³⁸ who have examined the effect of a vast number of surface finishes and have correlated the explosivity with the wettability of the surface by the coolant.

Experiments have been performed to study the effect of coolant viscosity on triggering.^{39,40} Experiments performed at Sandia by Nelson and Guay³⁹ showed that increasing the viscosity of the coolant could suppress explosions when tin drops were released into water. The viscosity was increased by using glycerol or cellulose gum. Also, a 50-kg scale test in which iron-alumina was dropped into a cellulose gum solution did not explode, where similar tests using untreated water did explode. It was postulated that the increased coolant viscosity prevents microjets of water from penetrating the melt droplets following film collapse so that rapid fragmentation cannot occur.

Single droplet experiments in which an iron-oxide melt was used have also been performed at the University of Wisconsin.⁴⁰ An external trigger of varying strength was applied as the droplets fell through the water pool. The water viscosity was changed from 0.04 to 0.24 kg m⁻¹ s⁻¹ by adding cellulose gum to the water. An increase in the viscosity was found to reduce considerably the likelihood of an explosion occurring.

More recently, workers at the Georgia Institute of Technology have investigated the effect of surfactants on the spontaneous interaction of tin droplets at 1073 K dropped into water.⁴¹ A variety of surfactants were used and were found to reduce the peak pressure of the spontaneous explosions that occurred. The paper also reviews other work in this area and notes the need for experiments using prototypical materials before any conclusions relevant to reactor safety applications can be drawn.

Although none of the practices previously described are applicable to the reactor application, they do highlight the sensitivity of the triggering process to the coolant and vessel properties.

TRIGGERING IN PROTOTYPIC INTEGRAL STEAM EXPLOSION EXPERIMENTS

This section reviews the triggering behavior observed in medium-scale experiments involving prototypic melts. The experimental programs performed at Winfrith, Argonne National Laboratory (ANL), SNL, JRC Ispra, and Japan Atomic Energy Research Institute (JAERI) are described in the following sections.

Experiments Performed at Winfrith

In all the experiments reported here, the melt used was a mixture of 81% (by weight) uranium dioxide and 19% molybdenum at an initial temperature of 3600 K.

The SUW Experiments. In the SUW experiments, thermite-generated melts in quantities of up to 24 kg were released under the surface of a pool of 1.5 tons of water within a pressure vessel [the Molten Fuel Test Facility (MFTF)].^{42,43} Two different modes of melt release were used. In the first, "free release," the melt was ejected from the charge container by the gases produced during the thermite burn and allowed to mix freely with all the water present. In the second, "restricted release," a catchpot was attached to the end of the charge container to restrict the amount of water mixing with the melt. The region above the water pool contained argon and steam, and it allowed the mixture to expand during the interaction. No external trigger was employed, but triggered explosions occurred when the heavy steel end-cap from the charge release mechanism hit the base of the pressure vessel. The initial conditions and results for the experiments are summarized in Table 2.

In the experiments, both triggered and spontaneous interactions were observed. At initial pressures above 0.1 MPa, only triggered interactions occurred. Thus these experiments appear to show that spontaneous interactions can be suppressed by a relatively small increase in the ambient pressure. It is not clear what the mechanism causing the spontaneous interactions was because they occurred as the melt was falling through the water pool (i.e., they were not caused by coolant entrapment against a solid surface). A feature of these experiments was the increased work yield with increased pressure. This was explained as improved mixing as the pressure increased because the calculated efficiency (based on the melt mass with a diameter less than 250 μ m) was independent of pressure.

The WUMT Experiments. In the WUMT experiments, the aim was to pour under gravity 24 kg of thermite-generated melt through a circular orifice into a tank of water contained within the MFTF pressure vessel.^{43,44} However, in most experiments the charge container was not fully vented, and the melt was ejected by the escaping gas stream. The test conditions are given in Table 3. No external triggers were applied.

Steam explosions occurred in only two of the nine experiments. In experiment 03, an explosion occurred some time before base contact. In this experiment, the water was subcooled by 80 K, and this large subcooling

Table 2 Test Conditions and Results for the SUW Experiments

Experiment	Release type ^a	Melt mass, kg	Pressure, MPa	Subcooling, K	Number of explosions ^b	Total work, MJ
01	F	24	0.1	78	3	0.233
02	F	24	0.4	87	0	
03	F	24	0.1	80	2	0.077
04	R	24	0.1	80	3	0.175
05	R	24	0.1	61	2	0.162
06	R	24	0.1	31	2	0.160
07	R	24	0.1	0	3	0.225
08	R	24	0.5	60	1	0.521
09	R	24	1.0	60	1	0.884
10	R	8	0.1	60	1	0.118
11	R	8	1.0	60	0	
12	R	8	1.0	60	1	c

^aF, free release; and R, restricted release.

^bNumber of different explosions that could be identified from the pressure data.

^cNot available.

Table 3 Test Conditions for the WUMT Experiments

Experiment	Pressure, MPa	Subcooling, K	Water depth, m	Vessel side, m	Drop height, m	Orifice diameter, m
01	0.1	0	0.5	0.4	0.6	0.07
03	0.1	80	0.5	0.6	0	0.07
04	0.1	0	0.5	0.6	0.015	0.04
05	0.1	0	0.2	0.6	0.6	0.10
06	0.1	0	0.5	0.2	0.6	0.04
07	0.1	0	0.2	0.2	0	0.10
08	1.0	0	0.5	0.6	0.6	0.10
09	0.1	0	0.5	0.6	0.6	0.10

may have been responsible for the spontaneous triggering observed. The other experiment in which an explosion occurred was 09, in which the water was saturated. The explosion occurred about 0.5 second after the melt had reached the base of the mixing vessel, which suggests that triggering occurred because of water entrapment in a melt pool. There was no evidence of surface interactions as the melt entered the water.

The MIXA Experiments. The MIXA experiments involved the release of approximately 3 kg of melt into a water pool.^{45,46} A droplet former ensured that the melt entered the water as a stream of droplets with a diameter of approximately 6 mm. "Skirts" of varying lengths were attached beneath the droplet former to control radial

spreading of the droplet stream. The mixing vessel was of square cross section with a diameter of 0.37 m; a water pool depth of 0.6 m was used. The initial pressure was 0.1 MPa in all the tests. The water was initially saturated in all but one experiment, in which the water was subcooled by 20 K. The parameters varied in the experiment, in addition to the water subcooling, were the length of the "skirt" (which controlled the extent of the radial spreading of the melt stream) and the melt flow rate. Details are given in Ref. 46.

No external triggers were applied, and no spontaneous explosions were observed in any of the experiments. This may have been because the mixtures were relatively weak (the melt fraction was typically 1%) or because the melt was dispersed, so entrapment could not occur.

Experiments Performed at Argonne National Laboratory

Data on triggering are available from two series of core melt-water mixing experiments performed at ANL.

The CWTI Series. Two experiments in the CWTI series were similar in character to the WUMT experiments described previously.⁴⁷ In these experiments, a 22-mm-diameter jet of corium [consisting of 60% (by weight) UO_2 , 16% ZrO_2 , and 24% stainless steel] at a temperature of 3080 K was injected into a 320-mm-deep water pool contained in a cylindrical vessel with a diameter of 210 mm. In experiment CWTI-9, the water pool was subcooled by 6 K, and in experiment CWTI-10 it was subcooled by 75 K. No external triggers were applied. There was no evidence of an explosive interaction in either of the experiments.

The CCM Series. A second series of mixing experiments, the CCM series, were performed to study coarse mixing and melt jet breakup.⁴⁸ These experiments used the same melt type as the CWTI experiments. The melt was poured under gravity into a cylindrical mixing vessel (with a diameter of 0.21 m in experiments 01 to 04 and 0.76 m in experiments 05 and 06). The initial pressure was 0.1 MPa in all the experiments. Table 4 gives a summary of the main parameters varied in the experiments. Again, no external triggers were applied.

In this experimental series, no steam explosions occurred, and there was no evidence of surface interactions.

Experiments Performed at Sandia National Laboratories

The largest experimental data base available is undoubtedly that from experiments performed at SNL. Results from 11 different test series provide useful information on triggering. These were performed using different melts and covered a variety of pressures, water subcoolings, and melt delivery modes. It should be noted

that experiments referred to as at ambient pressure were performed at the local ambient pressure of 0.083 MPa ($T_{\text{sat}} = 368 \text{ K}$).

The Open-Geometry Test Series. A total of 59 experiments were performed in the open-geometry test series to develop melt delivery techniques, to investigate triggering, and to access the work potential from steam explosions.

The first 48 experiments, reported by Buxton and Benedick,⁴⁹ used iron-alumina melt [composed of approximately 55% (by weight) iron and 45% alumina] in quantities ranging from 1 to 27 kg. In most tests the melt was released (using a trapdoor arrangement) under gravity into a steel tank (0.9 m in diameter and 1.1 m tall) containing water at ambient pressure and temperature. In a few tests the water was heated. Several parameters were varied in an attempt to determine what controls triggering and the explosion energy. The instrumentation was very limited, comprising a camera to view the melt entry and a pressure transducer hung in the water.

In general, explosions occurred without the use of external triggers. They could be suppressed in some cases by coating the tank with lard (as was observed by Long³⁷) but could be triggered in this case with a detonator. Wire screens were placed in the path of the melt to prefragment it, but this did not prevent triggering. Spontaneous triggering at the water surface was more likely at high-melt flow rates ($\sim 32 \text{ kg/s}$) than at low-melt flow rates ($\sim 4 \text{ kg/s}$), where triggering tended to occur at base contact.

The final 11 tests, reported by Buxton, Benedick, and Corradini,⁵⁰ were performed to investigate the interaction of corium A + R (containing 70% fully oxidized uranium and zirconium and 30% stainless steel) with water. Useful data were obtained only from experiments 54 to 59 because of difficulties with the melt release system, and only experiments 56 to 59 used corium. In the experiments, 10 to 20 kg of melt was released into a hemispherically shaped steel tank (diameter, 1.2 m;

Table 4 Test Conditions for the CCM Experiments

Experiment	Melt mass, kg	Jet diameter, mm	Water depth, m	Subcooling, K
01	2.15	25.4	1.06	43
02	11.15	20.0	0.63	1
03	3.34	25.4	1.10	0
04	9.24	50.8	1.07	37
05	11.34	50.8	1.07	45
06	12.79	50.8	1.07	0

height, 4.3 m) containing water. The tank was instrumented with four pressure transducers.

Two different triggers were used. The first was an SEI detonator, which contained 0.64 g of pentaerythritol tetranitrate (PETN) explosive. This produced a pressure pulse (at transducers located around the tank) with a rise time of 10 μ s and a peak pressure of 2 to 3 MPa with a very short duration. (The peak-pressure duration is of the thickness of a line on the pressure-time plot given, with most of the disturbance having an amplitude of about 0.5 MPa and lasting for about 4 ms.) The second was a detonator plus 0.76 m of Primacord containing 6 g of PETN mounted vertically in the tank. In this case, the peak pressure was 5 to 7 MPa with a slightly longer pulse width. The pressure signal was very "spiky" because of reflected pressure waves.

The two experiments used iron-alumina; in one case a spontaneous explosion occurred, and in the other case a spontaneous explosion was followed by a second, triggered by a detonator. An examination of the pressure traces from the detonator-triggered explosion suggested that it was more energetic. The experimenters noted that this was probably because the explosion occurred later and was better "tamped."

In two experiments in which corium was used, a detonator was used as a trigger. No explosion was observed in either case. It was believed that this was because of partial solidification of the melt and the presence of incondensable gases. To overcome these effects, the trigger energy was increased (by using the Primacord and reducing the delay time before firing the trigger). In one test the trigger did not work correctly, and no explosion occurred. In the other test, a mild explosion was observed. The experimenters estimated that the work done by the explosion was approximately one-third of the chemical energy of the detonator.

This series highlighted many of the considerations that needed to be addressed if simulants and external triggers were to be used to investigate steam explosions. In particular, the inhibition of triggering caused by melt solidification and the difficulty of using an external trigger were noted.

Experiments Performed in the EXO-FITS Facility. Five series of experiments performed in the EXO-FITS facility (i.e., in the open air) provide useful data on triggering.

1. *The MD series.* Data are available from the last 13 experiments in the MD series (experiments 7 to 19).⁵¹ (The first six were performed to develop the melt delivery system.) This series of experiments was used for

equipment development, and only cine data are available for most of the tests. In the experiments, 0.6 to 5.3 kg of iron-alumina melt was released as a coherent mass (rather than as narrow jets) into subcooled water (by 70 to 80 K). A water mass of 20 to 50 times the melt mass was contained in a Plexiglas tank. Of the 13 experiments performed, in 2 cases an explosion occurred before base contact (1 near the crucible lid, which fell in with the melt, and 1 near the water surface), in 6 cases a base-triggered explosion occurred, and in 5 cases no explosion occurred. Spontaneous explosions did not occur for melt masses below about 2 kg.

2. *The MDC series.* The MDC series of experiments was performed to investigate the properties of melt-water interactions involving corium melts.^{52,53} These experiments used melt masses ranging from 4 to 20 kg. A new crucible and delivery system was developed to deliver the melt reliably. Again a variety of behaviors ranging from benign interaction to energetic explosions was observed. A mass threshold of about 4 kg was found, below which spontaneous interactions were not observed. The conclusion was reached that the explosion behavior observed in these tests was similar in all aspects, including triggering, to that observed in the MD series.

3. *The MDF series.* In the MDF experiments, about 0.8 kg of iron oxide melt (generated using a thermite reaction between iron and potassium perchlorate) was released into a water chamber that was 0.3 m square and had a water depth of 0.3 m.⁵⁴ The water temperature is not specified in the reference but was probably ambient. The melt was observed to be dispersed before entering the water. No spontaneous explosions were observed, although a detonator was used to trigger an explosion in three of the tests. It was noted that in some tests individual droplets of melt exploded in a similar manner to that observed in single droplet tests, but they did not act as a trigger. These data are interesting because they again suggest that dispersion of the melt can inhibit base triggering, and they also highlight the fact that not all spontaneous triggers lead to a propagating event.

4. *The CM series.* The CM series of experiments was conducted in an instrumented Plexiglas tank and was designed to study coarse mixing.^{54,55} In these experiments iron-alumina melt was released into water from a melt crucible located above the mixing vessel. Because the experiments were designed to study the coarse mixing stage, no external triggers were applied. The experimental parameters are given in Table 5. The melt was released as a coherent mass from the crucible. In some

cases the bottom lid of the crucible was allowed to fall with the melt into the mixing vessel; in others it was not. The release type is shown in the table.

In these tests a very prompt interaction was observed between the melt and water at or very near the water surface. This interaction caused much of the melt to be ejected from the vessel, and in some cases the melt in the water was driven rapidly downward. Although these events were violent enough to disperse much of the melt, they were not violent enough to be characterized as steam explosions. It was postulated that this dispersion was caused by rapid steam and hydrogen generation as the melt contacted the water surface. Of the 10 tests carried out with nearly saturated water, only 3 (experiments 8, 9, and 10) resulted in steam explosions. The experimenters were unable to identify any of the parameters as controlling whether an explosion occurred or not, although they postulated that triggering may have been more likely in

cases where the lid was allowed to fall into the water (in both tests where the lid was allowed to fall in, explosions occurred). In contrast with the saturated case, both tests with subcooled water resulted in steam explosions. In these tests the surface interaction was weaker, and more melt participated in the explosions.

5. *The OM series.* The OM series of experiments was carried out to determine whether the behavior observed in the CM tests was caused by the use of a metallic melt.⁵⁵ A thermite-generated melt comprised of iron oxide was used instead of the iron-alumina used in the previous series. The initial conditions used in this test series are given in Table 6. Explosions were observed in all four tests. In the near-saturated water test (04), four spontaneous triggers were observed, two of which resulted in vigorous steam explosions. No surface interactions occurred in any of the tests. It is suggested that "this

Table 5 Test Conditions for the CM Test Series

Experiment	Melt mass, kg	Subcooling, K	Vessel side, m	Water depth, m	Drop height, m	Entry velocity, ms ⁻¹	Release type ^a
01	18.5	9	0.31	1.22	0.31	2.44	NL
02	18.0	4	0.31	1.22	0.31	2.44	NL
03	18.0	3	0.61	1.22	0.48	3.11	NL
04	18.9	3	0.61	0.61	1.12	4.60	NL
05	7.6	4	0.61	0.61	1.22	4.99	NL
06	4.0	3	0.61	0.61	1.33	4.99	NL
07	18.5	73	0.61	0.46	1.12	4.77	NL
08	18.6	2	0.61	0.61	0.44	3.08	L
09	18.6	3	0.61	0.61	0.44	3.06	L
10	18.4	1	0.61	0.31	1.14	4.60	NL
11	18.7	1	0.61	0.61	1.12	4.68	NL
12	18.5	69	0.61	0.31	1.82	5.89	L

^aNL, crucible lid did not fall with melt into the mixing vessel; and L, lid fell with the melt into the mixing vessel.

Table 6 Test Conditions for the OM Test Series

Experiment	Melt mass, kg	Subcooling, K	Vessel side, m	Water depth, m	Drop height, m	Entry velocity, ms ⁻¹	Release type ^a
01	Unknown	69	0.43	0.36	0.64	3.53	NL
02	9	69	0.53	0.36	0.64	3.83	NL
03	10	69	0.61	0.36	0.64	3.34	L
04	9	4	0.61	0.61	0.79	3.56	NL

^aNL, crucible lid did not fall with melt into the mixing vessel; and L, lid fell with the melt into the mixing vessel.

tends to support the hypothesis that hydrogen generation may have contributed to the surface eruptions.⁵⁵

Experiments in the FITS Facility. Five series of experiments performed at SNL in the Fully Instrumented Test Site (FITS) facility provide information on triggering. The FITS chamber is a pressure vessel approximately 3 m high and 1.5 m in diameter with an internal volume of 5.6 m³. In the experiments melt was released from a crucible into a square-section Plexiglas chamber contained within the pressure vessel. The facility is instrumented with high-speed cameras, pressure transducers, and gas-sampling equipment.

1. *The FITS-A series.* The FITS-A experiments were similar to the MD series but were performed inside the FITS vessel.^{51,56} They involved between 2 and 5 kg of iron-alumina melt released into water at ambient temperature. The experimental conditions are given in Table 7.

Each experiment behaved differently. In 1A there was a mild interaction, in 2A a surface explosion occurred, in 3A an explosion was triggered when the mixture front was halfway down the vessel, in 4A no explosion occurred, and in 5A an explosion was triggered with a detonator (containing 0.64 g of PETN) when the melt was beginning to collect on the vessel base.

2. *The FITS-B series.* The FITS-B series of experiments was performed to study the effect of initial melt/water mass ratio and geometry on the explosivity of an iron-alumina melt in water.^{52,57,58} The experiments were carried out at ambient pressure. The initial conditions used in this series are given in Table 8. No external triggers were applied.

The results can be summarized as follows:

- A surface interaction or explosion followed by a base-triggered explosion occurred in experiments 1B, 3B,

Table 7 Test Conditions for the FITS-A Test Series

Experiment	Melt mass, kg	Ambient pressure, MPa	Water subcooling, K	Vessel side, m	Water depth, m	Entry velocity, ms ⁻¹
1A	1.94	0.083	85	0.46	0.43	1.9
2A	2.87	0.083	82	0.53	0.53	2.9
3A	5.3	0.083	81	0.61	0.61	5.3
4A	4.3	1.1	80	0.61	0.61	4.3
5A	5.4	1.1	81	0.61	0.61	5.4

Table 8 Test Conditions for the FITS-B Test Series

Experiment	Melt mass, kg	Water subcooling, K	Vessel side, m	Water depth, m	Entry velocity, ms ⁻¹
1B	18.7	70	0.61	0.61	5.4
2B	18.6	70	0.61	0.30	6.0
3B	18.6	67	0.43	0.30	6.0
4B	18.7	69	0.61	0.61	6.8
5B	14.5	1	0.46	0.37	Unknown
6B	18.7	1	0.46	0.30	7.2
7B	18.7	78	0.43	0.15	7.4
7BR ^a	18.7	79	0.43	0.15	6.8
8B	18.7	81	0.61	0.77	6.5
9B	18.7	80	0.61	0.46	7.0

^aPerformed in EXO-FITS facility to allow improved photography.

4B, 7BR, and 8B. In experiment 3B, the surface interaction was weak; in the other tests it was classified as an explosion, as there was clear evidence (from the pressure records) of shock pressurization of the FITS chamber. The Plexiglas chamber failed in the experiments in which a surface explosion occurred.

- A single surface interaction occurred in experiment 2B.
- A single base-triggered interaction occurred in experiment 9B, and a single explosion occurred in experiment 7B (its location was unknown because of a camera failure).
- No explosions occurred in experiments 5B and 6B. In experiment 5B the cameras did not work, and in experiment 6B there were four local events that did not propagate.

In this series triggering was again suppressed by reducing the water subcooling (no explosions occurred in near-saturated water). There are no obvious trends for the effect of water depth or mixing vessel diameter on the likelihood of triggering. A second explosion occurred more often in a deep water pool, most likely because of water depletion in the first event in a shallow pool. The experiments also suggested that for small water depths the explosion efficiency is reduced because of a lack of tamping. This effect also depends on the triggering time because early triggers give explosions, which are poorly tamped, at the top of the water pool; triggering soon after the melt reaches the vessel base leads to well-tamped explosions in deep-water pools, particularly at high subcooling, so that the steam fraction in the overlying water slug is low.

When the triggers were recorded by the cameras, it was observed that "the triggers appeared as rather complicated wave-like phenomena in the water surrounding the melt mixture." Triggers were observed "at or near the

water surface; at or near the water chamber base or side walls; on occasions, at all of these locations." These observations clearly highlight the complex and random nature of the triggering event.

3. *The FITS-C series.* The FITS-C experiments, summarized in Refs. 59 and 60, used 10 to 20 kg of either iron–alumina thermite or corium. They were performed to study the effect of melt composition on explosivity, hydrogen production, and debris formation. A nitrogen atmosphere was used so that gas samples could be taken to determine the hydrogen content. Table 9 gives the initial conditions and results for the five tests in this series. Again, no external triggers were applied (external triggers were planned for experiments 4C and 5C but did not operate).

In experiments 1C and 2C, spontaneous surface-triggered explosions occurred. Also, a weak triggered explosion in experiment 2C occurred as residual melt reached the base of the vessel. No explosions were observed in the other experiments. In experiment 3C the melt was dispersed in the form of 10- to 20-mm-diameter droplets when it reached the water surface. There was some evidence from the form of the debris that the melt was partially solidified when it reached the vessel base in experiments 2C and 3C. It was postulated that this was caused by the use of a nitrogen atmosphere. Concerns over the melt behavior caused the experimenters to return to the use of iron–alumina for the remainder of the program. In experiment 4C much of the data was lost, so the experiment was repeated in 5C. In this case the melt was again dispersed at delivery, and no explosion resulted.

Although difficult to interpret, these experiments suggest that both low-melt superheat and dispersion of the melt before contact with the water reduce the likelihood of triggering.

Table 9 Test Conditions and Results for the FITS-C Experiments^a

Experiment	Melt type ^b	Melt mass, kg	Water subcooling, K	Ambient pressure, MPa	Water chamber side, m	Water depth, m	Melt velocity, ms ⁻¹	Event ^c
1C	IA	17.1	69	0.083	0.61	0.31	5.6	SE
2C	C	16.0	72	0.083	0.61	0.61	6.6	Weak SE
3C	C	11.5	68	0.083	0.53	0.38	6.0	B
4C	IA	19.0	67	0.55	0.61	0.31	6.0	B
5C	IA	19.6	69	0.52	0.61	0.31	6.0	B

^aSource: Ref. 60.

^bC, corium; and IA, iron–alumina.

^cSE, steam explosion; and B, benign interaction.

4. *The FITS-D series.* The FITS-D series of experiments⁶¹ was performed to clarify some of the issues raised in the FITS-C experiments. In these experiments about 20 kg of iron–alumina thermite-generated melt was released under gravity into water. No external triggers were applied in these tests. The experimental conditions are given in Table 10.

The only explosive interaction in this series occurred in experiment 5D. In this test the water was highly subcooled, and the explosion occurred approximately 53 ms after melt–water contact. It was one of the most violent explosions ever observed in the FITS vessel. Examination of the high-speed motion picture data showed that the explosion was the result of two separate propagation events, separated in time by only 3 ms. Nonexplosive interactions were observed in experiments 0D and 8D. In these experiments some material was expelled from the mixing vessel, but this was due to a rapid steam flow on a nonexplosive time scale. The remaining experiments, which had initial pressures in the range 0.7 to 1.1 MPa, resulted in a benign interaction in which the melt mixed with water, failed to trigger, and then agglomerated on the base of the vessel.

5. *The FITS-G series.* The FITS-G experiments, which were carried out between the FITS-A and FITS-B

series, were performed to examine steam production rates for nonexplosive corium–water interactions.^{53,62} To avoid an explosion, the experimenters used (1) near-saturated water, (2) a high entry velocity (to entrain air with the melt and to break up the melt into a dispersion of droplets), and (3) a thick Lucite base (which decomposes to produce gas, which suppresses triggering).

This procedure worked. In experiment 1G, 20.4 kg of melt was released into 44.4 kg of near-saturated water, and in experiment 2G, 13.6 kg of melt was released into 110 kg of saturated water. In both cases the interaction was nonexplosive, but there was vigorous steam production.

Experiments Performed at JRC Ispra

The FARO Quenching Tests. A series of experiments is being performed in the FARO facility at JRC Ispra to investigate the quenching of large masses of corium in water.⁶³ The melt is composed of 80% UO₂ (by weight) and 20% ZrO₂. The apparatus consists of a melt generator and an interaction vessel with a volume of 1.5 m³, which can withstand a pressure of 10 MPa at a temperature of 673 K. The water pool can be up to 2.5 m deep, and the vessel diameter is 0.71 m. To date, two experiments have been performed, the main features of which are given in Table 11.

Table 10 Test Conditions and Results for the FITS-D Experiments^a

Experiment	Melt mass, kg	Water subcooling, K	Ambient pressure, MPa	Water chamber side, m	Water depth, m	Melt velocity, ms ⁻¹	Event ^b
0D	17.8	0	0.085	0.61	0.51	5.9	E
2D	19.0	169	1.1	0.38	0.66	7.3	B
2DR	18.7	158	1.1	0.38	0.66	7.3	B
3D	18.9	37	0.7	0.76	0.15	5.7	B
5D	19.2	83	0.083	0.76	0.66	5.7	SE
8D	19.5	0	0.083	0.38	0.15	6.9	E

^aSource: Ref. 61.

^bSE, steam explosion; E, eruption, nonexplosive; and B, benign interaction.

Table 11 Test Conditions for the FARO Experiments

Experiment	Melt mass, kg	Melt temperature, K	Melt flow rate, kg/s	Ambient pressure, ^a MPa	Water subcooling (top), K	Water subcooling (bottom), K
Scoping Test (ST)	18	2923	64	5.4	2	38
Quenching Test 2 (QT2)	44	3023	119	6.1	12	20

^aAt melt/water contact.

No triggers were applied in the experiments, and no explosions were observed. It is noteworthy that in the scoping test a heater failed at the bottom of the vessel and that the lower 250 mm of the water pool was highly subcooled. No explosion occurred in this experiment or in quenching test 2, however. These results provide additional data on the effect of pressure on triggering because in both tests the pressure was above 5 MPa. Note that in the tests about 30% of the melt arrived at the base of the vessel in a molten state and produced an agglomerate, but water entrapment did not produce a trigger.

Further experiments in this series, in which increased melt masses and melts containing zirconium will be used, are planned.

The KROTOS Tests. The KROTOS facility has been used to examine steam explosion propagation for a number of years. Initially, the experiments involved molten salt or molten tin and water. More recently, experiments have been performed with aluminum oxide and water.⁶⁴ In the experiments, about 1.5 kg of melt at an initial temperature of about 2600 K was poured into a test section with a diameter of 0.4 m and a height of 2.2 m. The test section was instrumented with pressure transducers, thermocouples (to determine the melt location), and a level swell meter. To date, all the experiments have been performed at ambient pressure, and the main variable has been the water subcooling. A water depth of 1.1 m was used in all the tests. The main features of the tests are given in Table 12.

The experiments are described here in order of increasing water subcooling. In experiment 27, with nearly saturated water, there was a long period of steaming lasting several minutes. In experiment 28, a steam explosion was triggered for almost the same conditions as experiment 27 using the strong gas trigger (15 cm³ of argon at 8.5 MPa

released at the base of the vessel), which was activated at a preset time after melt release. A propagating interaction was observed. In experiment 26, the trigger was activated when the melt had penetrated only a small way into the water pool. Nevertheless, an explosion was triggered. In experiment 29, a spontaneous trigger occurred when the melt was still 150 mm from the base of the interaction vessel. The explosion was very strong and produced pressures of the order of 100 MPa. Experiment 30 was a repeat of experiment 29 but with some modifications to try to avoid the occurrence of a spontaneous interaction. (A tin membrane used to slow the melt at entry was removed to eliminate the possibility of a trigger from a tin-water interaction, and a Plexiglas liner was inserted in the interaction vessel.) An explosion was again spontaneously triggered, this time when the melt front was about halfway through the water pool. Again, pressures in excess of 100 MPa were recorded, and the apparatus received significant damage.

The experimenters have used these data to highlight the effect of subcooling on the triggering process. Future tests are planned using 5 kg of melt with the same melt composition as that used in the FARO tests.

Experiments Performed at JAERI

Workers at JAERI have performed a series of experiments to investigate the interaction of melt poured into water.⁶⁵ These experiments are referred to as the Melt Drop Steam Explosion Experiments (STX). The experiments were performed in a model containment (called the ALPHA facility), which has a diameter of 3.9 m and is 5.7 m high, enclosing a volume of 50 m³. It is possible to pressurize the system, with nitrogen, to examine the effect of pressure. Melt was produced in quantities of 10 or 20 kg from the thermite reaction of iron oxide with aluminum. The initial melt temperature was between 2700 and 3450 K. The melt was poured through a 200-mm-diameter orifice and fell through a height of 3.5 m before contacting the water with a speed of approximately 8 ms⁻¹.

Two different interaction vessels were used. One was made of steel and was cylindrical with a diameter of 1 m and a height of 1.2 m. The other was made of acrylic and was of square section with length of side 0.88 m and a height of 1.2 m. The water depth was 1 m in all the tests. In some tests, a grid of 2-mm-diameter steel wires with a pitch of 25 mm was placed 100 mm above the water surface to predisperse the melt before it entered the water. The main features of the experimental series are given in Table 13. No external triggers were used in any of the tests.

Table 12 Test Conditions and Results for the KROTOS Experiments

Experiment	Melt mass, kg	Water subcooling, K	Event ^a
27	1.0	10	B
28	1.22	13	TSE
26	<1.0	40	TSE
29	1.5	80	SSE
30	1.5	80	SSE

^aTSE, triggered steam explosion; SSE, spontaneous steam explosion; and B, benign interaction.

Table 13 Test Conditions and Results for the STX Experiments^a

Experiment	Melt mass, kg	Pressure, MPa	Water subcooling, K	Vessel ^b	Grid ^c	Event ^d
02	20	0.1	84	S	N	SE
03	20	0.1	81	S	N	SE
05	20	0.1	73	A	N	SE
09	20	0.1	84	A	N	SE
01	10	0.1	80	S	N	N
10	10	0.1	76	A	N	SE
08	20	1.6	186	A	N	N
06	20	0.1	75	A	Y	N
11	20	0.1	83	A	B	M

^aSource: Ref. 65.^bS, steel; and A, acrylic.^cN, no grid; Y, grid; and B, grid broken locally.^dSE, steam explosion; N, no explosion; and M, mild explosion.

As far as triggering is concerned, there seem to be three important points to note from the results of this test series. First, reducing the melt mass from 20 to 10 kg appeared to reduce the likelihood of a spontaneous interaction. Second, increasing the pressure to 1.6 MPa appeared to suppress an explosion that occurred in four similar tests performed at 0.1 MPa. Third, the dispersion device appeared to suppress an explosion for a condition in which an explosion readily occurred without a dispersion device. The authors note that the effect of the device is not clear. They postulate that it could have prevented an explosion because of the greater water depletion in the mixture or because the increased amount of air that would have been entrained with the melt enhanced the vapor film stability.

IMPLICATIONS FOR THE USE OF TRIGGERING ARGUMENTS IN ASSESSMENTS

In this section an attempt is made to summarize the data from the experimental and theoretical work described previously. These data are then used to draw some conclusions concerning the use of triggering arguments in steam explosion assessments. As an example, the approach used in quantification of the steam-explosion-induced containment failure probability for the Sizewell B PWR¹² is presented.

Summary of the Available Data

Before drawing conclusions from the data, two questions should be addressed. The first question concerns the

relevance of the data from systems other than the corium–water system. Clearly, some of the triggering mechanisms discussed previously do not apply in this system. For example, vapor film collapse caused by cooling of the melt leading to spontaneous triggering is unlikely to be important in the reactor application because melt near the surface of a droplet will freeze long before the minimum film boiling temperature is reached. Thus data on spontaneous triggering from nonprototypical systems must be treated with caution. With this provision, however, it is still possible to use the data on, for example, molten salt–water explosions to learn something about the effect of pressure on triggering.

The second question concerns the effect of scale. Most triggering experiments have used gram quantities of melt, and most integral tests have used kilogram quantities. In the reactor application, ton quantities of melt must participate for the vessel integrity to be threatened. Thus one is tempted to dismiss the current data base as irrelevant. This would be a far too simplistic view, however. For ton quantities of melt to explode, a progression through the stages of mixing, triggering, and propagation is required. If we look at what we know about the triggering process, it is clear that this is a localized phenomenon, and it will occur in some small region of the mixture. Thus information derived from integral tests in which the mixing zone has a dimension of the order of 0.1 m is relevant. This argument is applicable if the melt stream entering the water has broken up into droplets. If it has not, then increasing the mass scale (or, more likely, the mass flow rate) does increase the likelihood that water will be trapped by falling melt and entrapment triggering will

occur. However, this situation is likely to lead to a much lower energy release than if the system were premixed. In summary, although scale has an effect on triggering, the localized nature of the triggering event means that data from experiments using much less than ton quantities of melt can be used to examine what factors affect the likelihood that triggering will occur in any given situation.

Keeping these issues in mind, the following conclusions and comments can be drawn from the available data:

- There are no developed and validated triggering models that can be used with any degree of confidence.
- The evidence from model predictions is that triggering becomes more difficult at higher pressure and for higher melt temperatures. As the pressure increases, the vapor mass and energy densities increase and the latent heat of vaporization decreases so that it becomes more difficult to compress the film, more difficult to condense the vapor, and easier to evaporate the leading edge of the water slug.
- The presence of a permanent gas can affect the triggering process. Small quantities of gas inhibit triggering, whereas rapid gas evolution can lead to spontaneous explosions.
- Experimental data show very clearly the random nature of the triggering process.
- Explosions can be triggered as the melt enters the water pool, as it is falling, upon base contact, or after melt has collected on the base of the mixing vessel. Explosions frequently occur without an applied external trigger.
- The spontaneous explosions that occur when melt contacts the water can be suppressed by a small increase in the ambient pressure (as little as 0.5 to 1.0 MPa is often sufficient).
- There is no clear evidence for a triggered explosion occurring at pressures above about 3 MPa. An explosion was triggered at 5.8 MPa in the HPTR experiments, but this involved the injection of a slug of cold water into the mixture.
- Explosions are much more likely to occur in subcooled conditions compared with saturated conditions.
- There is considerable evidence that if the melt is predisposed, it is much less likely that an explosion will trigger.
- There is evidence that if the melt has a low superheat, partial solidification during the melt-water interaction can inhibit triggering.

Possible Trigger Magnitudes

It seems that there are two obvious means by which an explosion could be triggered in the in-vessel situation in

the absence of operator action. (Note that if water is injected during the melt-water interaction, this could act as a trigger by collapsing the vapor film caused by the flow by increasing the water subcooling.) The first of these is by the entrapment of water within the body of the melt or against a solid surface. If the melt heats the water until homogeneous nucleation occurs, then a pressure of about 9.8 MPa would be generated in the low-pressure case (0.1 MPa) [because the pressure at the homogeneous nucleation temperature is given by $p_{hn} = p_{sat}(T_{hn})$ and the homogeneous nucleation temperature T_{hn} is given by $T_{hn} \approx 0.9T_{crit}$ for a pressure of 0.1 MPa].¹ This pressure is clearly sufficient to lead to vapor film collapse. At higher ambient pressures the homogeneous nucleation temperature is more difficult to calculate, but it increases with pressure. The value of p_{hn} must be below the critical pressure of 22 MPa. (In reality, heterogeneous nucleation is likely to occur at a temperature below the homogeneous nucleation temperature because of the presence of crud at the melt-water interface or dissolved gases in the water.) Thus at higher pressures this form of triggering is likely to be much less effective because it is the difference in pressure that causes vapor film collapse. This result agrees with physical intuition because, as the pressure is increased, the volume change upon vaporization decreases, and thus the disruptive force must be reduced.

The lower head of a reactor contains much internal structure, and there are many places where melt could collect and trap water. Thus the mechanism described is likely to be more common in the reactor application than it is in experimental studies in structure-free vessels.

The second means of triggering is by a mechanical impact, for example, by a falling steel structure. When such a missile hits a fixed structure, a hammer pressure is developed. If a missile with velocity v and acoustic impedance $(\rho c)_{inc}$ (where ρ is the material density and c is the speed of sound) is brought to rest by a fixed object with acoustic impedance $(\rho c)_{tar}$, then, if normal incidence is assumed, the hammer pressure is given by⁶⁶

$$\Delta p_{ham} = (\rho c)_{inc} v \left[\frac{(\rho c)_{tar}}{(\rho c)_{inc} + (\rho c)_{tar}} \right] \quad (2)$$

In the case where the falling object and the target have the same material properties, this reduces to

$$\Delta p_{ham} = \frac{1}{2} (\rho c)_{inc} v \quad (3)$$

For a steel component, $\rho \approx 8000 \text{ kg m}^{-3}$ and $c \approx 6000 \text{ ms}^{-1}$, so $\Delta p_{\text{ham}} \approx 24v \text{ MPa}$ if the velocity v is in ms^{-1} . Thus a steel object falling at a few meters per second could generate a considerable hammer pressure. The duration of this pressure pulse would be of the order of a typical length scale of the object divided by the speed of sound in steel. This gives a typical pulse duration of 0.17 ms/m . Thus, although the hammer pressure is high, its duration is fairly short. Nevertheless, according to the triggering classification proposed by Berman and Beck,²⁸ a 1-m-long object falling at a few meters per second would create a strong trigger.

These estimates should be compared with the characteristics of triggers used in previous steam explosion experiments. The detonator and detonator-Primacord arrangements used at Sandia gave peak pressures of the order of 5 MPa with very short durations, followed by lower pressure disturbances of the order of 0.5 MPa, lasting for a few milliseconds. These characteristics are not dissimilar from the estimates just given for triggers that could occur in a reactor accident.

Previous Use of Triggering Arguments

In Briggs' conservative assessment, the probability of an effective trigger occurring at low pressure was set at 0.6, and at high pressure it was set at 0.2 (Ref. 67). He used the term "effective trigger" to mean triggering an explosion when a significant mass of melt is in contact with water. These probabilities were chosen on the basis that early triggering is quite likely at low pressure, so the probability of there being an effective trigger once a significant mass of melt has mixed is less than unity because early triggering may have already occurred and dispersed the melt and water. At high pressure the relatively low probability of triggering was based on the view that experiments and theory indicated that triggering is unlikely at high pressure.

A paper by Bankoff and Yang⁶⁸ contains a discussion of the use of triggering arguments in developing a steam-explosion-induced vessel failure probability. They concluded that the probability is "virtually nil" on the basis of the following:

1. The high probability of early triggering, which could cause a premature explosion and vessel pressurization.
2. The impossibility of subsequently triggering an explosion once a critical pressure threshold has been exceeded.

Their argument is based on the observation that interactions are often triggered when melt comes into contact

with structures and that the lower head of a reactor contains "thousands of such contact points" and an explosion will trigger with a probability of "virtually one." At low ambient pressures they expect this explosion to be too small to threaten the vessel integrity because triggering will occur before a significant fraction of melt has entered the lower head. However, "it would mix large quantities of water with the remaining melt in the core and thus cause the vessel to pressurize and therefore to inhibit any further explosion." They believe a reasonable upper bound on the pressure at which an explosion can occur is 6.7 MPa (on the basis of fragmentation modeling work performed by Buchanan).⁶⁹ Thus they conclude that, "if the initial pressure is high, there will be no explosion. If it is only a few bars, an explosion can proceed but will pressurize the vessel to prevent further explosions. The intermediate initial pressure range of 20 to 50 bars has not been explored, but such explosions are weak."⁶⁹

This argument cannot be sustained. It is not clear why the small, initial explosion mixes melt and water sufficiently well to generate sufficient steam to pressurize the vessel without a second (larger) explosion occurring. The arguments about the effect of pressure on triggering are based on predictions from a fragmentation model that has not been verified experimentally. The model, developed by Buchanan,⁶⁹ is based on the idea that melt fragmentation occurs by coolant jet penetration into the body of the melt. This coolant then vaporizes (at either the heterogeneous or homogeneous nucleation temperature), the pressure rises, and a bubble of vapor is formed. As this bubble of vapor expands into the subcooled coolant around it, it condenses and causes a coolant jet to penetrate the melt, and the whole process repeats itself cyclically. Buchanan found that successive bubbles would have an increased pressure if the external pressure was below 1.3 MPa if homogeneous nucleation was the means by which the coolant jet was vaporizing or 6.8 MPa if heterogeneous nucleation was occurring. This result has been used to suggest that explosions between molten lava and water are suppressed for water depths greater than a certain value.⁷⁰ There is no solid evidence to justify its use. In addition, experiment HPTR05 came very close to invalidating their upper bound for triggering.

In the deliberations of the Steam Explosion Review Group (SERG), most participants thought that triggering at low pressure was relatively likely and becomes harder with increased pressure.⁷¹ However, various members gave different weight to the use of triggering arguments in their evaluations of the α -mode failure probability, mainly because they were using different rules for assigning probabilities.

In other published assessments of the α -mode probability, triggering arguments are not explicitly used. For example, Theofanous et al.^{72,73} have performed a comprehensive assessment for the low-pressure case without using triggering arguments.

Use of Triggering Arguments in the Sizewell B Steam Explosion Assessment

As an example of how the data from this review can be used in a probabilistic manner, the use of triggering arguments in the Sizewell B steam explosion assessment is described.¹² In the assessment it was decided to split the α -mode failure process into a number of discrete events and to use probability distributions to represent the range of possibilities and uncertainties in key quantities, such as the time that the melt first contacts the vessel base or the explosion conversion efficiency. A Monte Carlo approach is then used to sample from all the distributions to generate the final probability.

Triggering was treated by sampling from a cumulative probability distribution that represented the likelihood of triggering: (1) in the early stages of the interaction, (2) before base contact, (3) before all the melt reached the base of the vessel, and (4) after all the melt was in a pool at the base of the vessel. The probabilities of triggering in the different stages are given in Table 14, where p_1 is the probability that triggering occurs in the initial interaction, p_2 is the probability that it occurs before the melt first contacts the base of the vessel, p_3 is the probability that it occurs before all the melt has settled on the base of the vessel, and p_4 is the probability that it occurs at all. Thus $p_4 - p_3$ is the probability of a stratified explosion being triggered once all the melt is in a pool, whereas $1 - p_4$ is the probability that there is no trigger.

The data given in Table 14 show that at low pressure it was concluded that the probability of triggering before base contact is 0.3, that the probability of no trigger is 0.3, and that the probability of a trigger occurring while

the melt was collecting on the base of the vessel is 0.3. This choice was based on the observations from the experimental data together with consideration of the distribution of internal structure within the lower head of the vessel.

At higher pressures the probability of an interaction during initial melt-water contact was judged to be very low. In addition, the probability of a trigger occurring at all was reduced from 0.7 at 0.1 MPa to 0.2 at 6 MPa and to 0.1 at 15 MPa. These values are conservative (i.e., a strong claim for the suppression of triggering at high pressure was not made because of the lack of relevant experimental data).

CONCLUSIONS

Following a review of the available experimental data and of the modeling work available to date, the following conclusions can be drawn with regard to the use of triggering arguments in steam explosion assessments:

- It is not possible to claim a significant amount for the reduction in the α -mode failure probability because of early triggering. Such a claim is not supported by the experimental data.
- Although triggering is likely at a pressure of 0.1 MPa, it is not as likely as some workers have claimed. A small increase in pressure, of only a few bars, can suppress the spontaneous interactions observed in some experiments.
- Triggering is more difficult at higher pressure, as evidenced by the results from the Ispra molten salt tests, the FARO experiments, the FITS tests, the STX experiments performed at JAERI, and the TMI-2 accident. There is no reason to believe that it is impossible at a pressure of about 6 MPa. However, the required trigger magnitude may be much larger than that which is available in a reactor accident.
- If mixing occurs in subcooled water, as is likely in ex-vessel melt-water interactions, the likelihood of triggering is increased significantly.
- Any estimates of triggering probability are subjective, but there is a relatively large pool of data from experiments of the order of 10 kg of prototypical melt on which to base this judgment.

RECOMMENDATIONS

It is clear from this review that one of the crucial areas where more data would be useful is that of triggering at

Table 14 Pressure Dependence of Parameters in the Triggering Distribution used in the Sizewell B Assessment^a

Pressure, MPa	p_1	p_2	p_3	p_4
0.1	0.2	0.3	0.6	0.7
6.0	0.02	0.05	0.15	0.2
15.0	0.01	0.04	0.08	0.1

high pressure. A systematic study of the effect of pressure on triggering in experiments using prototypical material and a reasonable mass of melt is desirable. The ALPHA facility in Japan would seem ideally suited to this task. In such experiments it would be important to use realistic triggers, such as a mechanical impact, of the magnitude possible in a severe reactor accident. In addition, because of the importance of subcooling in determining the likelihood of triggering, it is essential that this parameter is also controlled. For in-vessel applications, experiments with a subcooling of about 10 K would be appropriate.

Experiments that examine the role of structures on triggering would also be useful. Again, it would be important to choose prototypical conditions and to use prototypical structures. For example, for a PWR, the effect of below-core support plates and instrumentation could be investigated relatively easily by putting structures within the mixing vessel. The experiments would have to be carefully designed to ensure that the melt mass and geometry represent a local region within the reactor vessel.

For ex-vessel studies, the fact that experimental data suggest that partial solidification can inhibit triggering is obviously of interest. A systematic study of the effect of melt superheat on spontaneous triggering using a prototypical melt would be useful.

As far as suppression of explosions is concerned, it is clear that more research is needed before any of the proposed methods can be adopted. The effect of additives in the water is poorly understood. The method most promising appears to be the use of a grid to predisperse the melt. This method of suppression is most probably connected with the effect of partial solidification and could be investigated at the same time as this variable.

It is not clear if additional model development would be very useful because the detailed mechanisms that occur during triggering are unlikely to be known in sufficient detail to design a comprehensive model. In addition, validation of such a model is likely to be impossible.

Most progress in this area is likely to come from using the available data, together with that from new and continuing experimental series (such as FARO and KROTOS), to develop an enlarged data base on triggering. Additional experiments to address triggering in other contact modes may be desirable as new areas of interest develop or new reactor types are considered.

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Analysis and Modeling of Flow-Blockage-Induced Steam Explosion Events in the High-Flux Isotope Reactor

By R. P. Taleyarkhan, V. Georgevich, C. W. Nestor, U. Gat, B. L. Lepard,
D. H. Cook, J. Freels, S. J. Chang, C. Luttrell, R. C. Gwaltney,
and J. Kirkpatrick^a

Abstract: *This article provides a perspective overview of the analysis and modeling work done to evaluate the threat from steam explosion loads in the High-Flux Isotope Reactor (HFIR) during flow blockage events. The overall work scope included modeling and analysis of core-melt initiation, melt propagation, bounding and best-estimate steam explosion energetics, vessel failure from fracture, bolts failure from exceedance of elastic limits, and, finally, missile evolution and transport. Aluminum ignition was neglected. Evaluations indicated that a thermally driven steam explosion with more than 65 MJ of energy insertion in the core region over several milliseconds would be needed to cause a sufficiently energetic missile with a capacity to cause early confinement failure. This amounts to about 65% of the HFIR core mass melting and participating in a steam explosion. Conservative melt propagation analyses have indicated that at most only 24% of the HFIR core mass could melt during flow blockage events under full-power conditions. Therefore it is judged that the HFIR vessel and top head structure will be able to withstand loads generated from thermally driven steam explosions initiated by any credible flow blockage event. A substantial margin to safety was demonstrated.*

The U.S. Department of Energy's (DOE's) High-Flux Isotope Reactor (HFIR) is an 85-MW research reactor located at Oak Ridge National Laboratory (ORNL). Figures 1a and 1b are a schematic representation of the HFIR and an illustration of the HFIR core, respectively. The HFIR uses highly enriched U_3O_8 -Al fuel with aluminum cladding. Various important design and operating

parameters of the HFIR are (1) flux trap cylindrical annulus geometry core 0.5 m in height, (2) plate-type fuel geometry with plate and coolant gap thicknesses of 1.27 mm each, (3) rated core pressure drop and coolant velocities in the core of about 0.7 MPa and 15.2 m/s, (4) rated core power density of about 1.8 MW/L, and (5) a 2.6-m-diameter pressure vessel located at the bottom of a 4.3-m-deep reactor pool. As can be expected, such features as a very high power density make the HFIR particularly susceptible to loss of pressure and flow transients. A level-1 probabilistic risk analysis (PRA) study¹ conducted for the HFIR has concluded that core damage frequency from internal events is dominated by flow blockage events. A large enough flow blockage may cause rapid fuel melting under full power conditions, which may then lead to steam explosions. Therefore a comprehensive study was undertaken to evaluate the threat to vessel and confinement integrity from steam explosion loads during flow blockage events.² Results of this work have been included in the recently completed HFIR Safety Analysis Report (SAR).

The basic approach followed is explained in Fig. 2. The approach consisted of evaluating what fraction of the HFIR core could realistically melt from the occurrence of small or large core flow blockages. Simultaneously, a comprehensive analytical framework was developed to evaluate the energetics of a resulting steam explosion coupled with an analysis of HFIR vessel and top head bolt failure characteristics. A key assumption for modeling and analysis of steam explosion energetics involved neglecting chemical energy sources from aluminum ignition in water. It is then shown that, for realistic upper-bound values of core melting, the resulting steam explosion loads are tolerable in the sense that they do not compromise the integrity of the reactor vessel or the top head bolts. Thereafter the margin to safety is evaluated by

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Note: Drawing not to scale

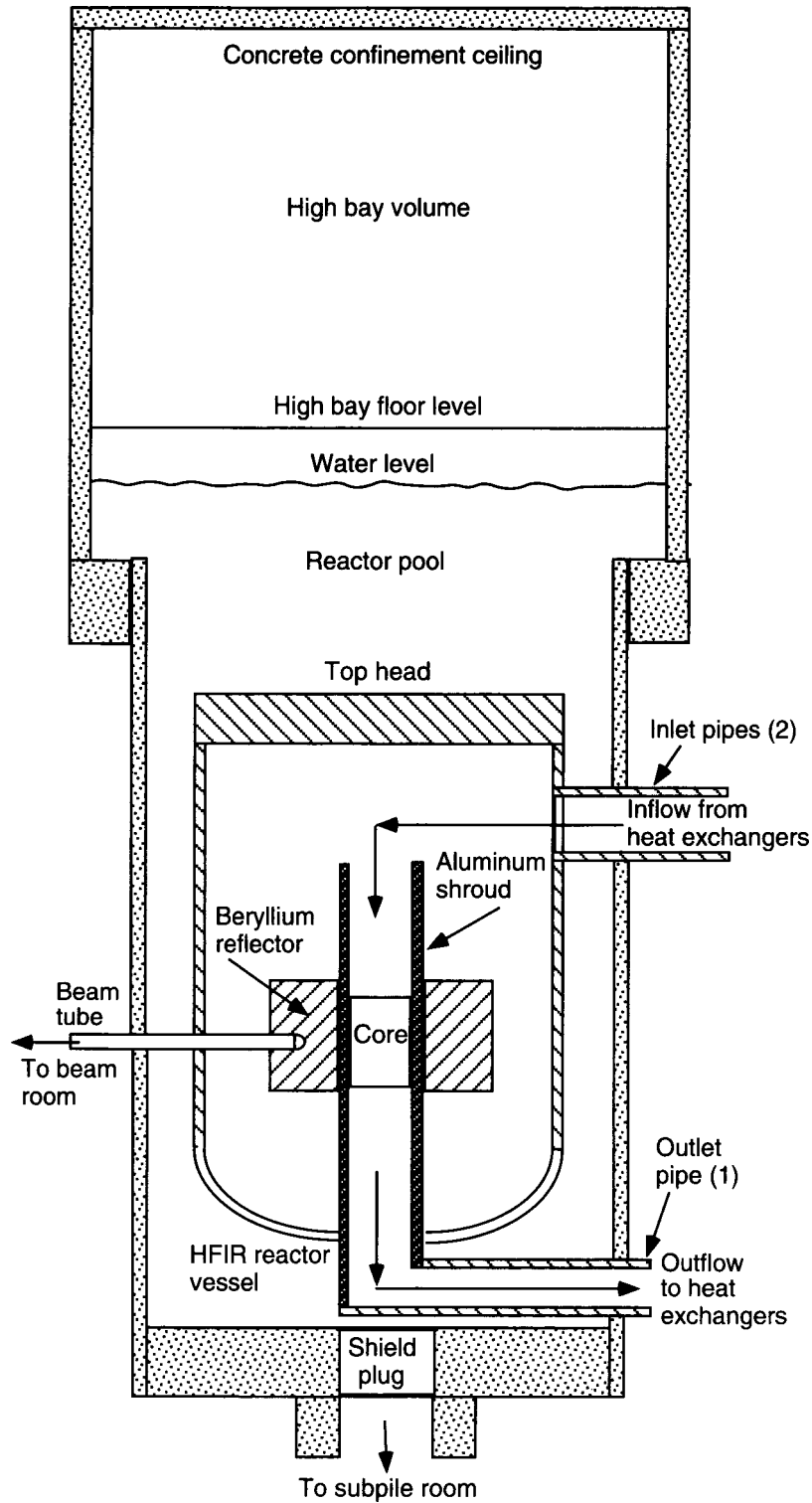


Fig. 1a Schematic of High-Flux Isotope Reactor (HFIR) system.

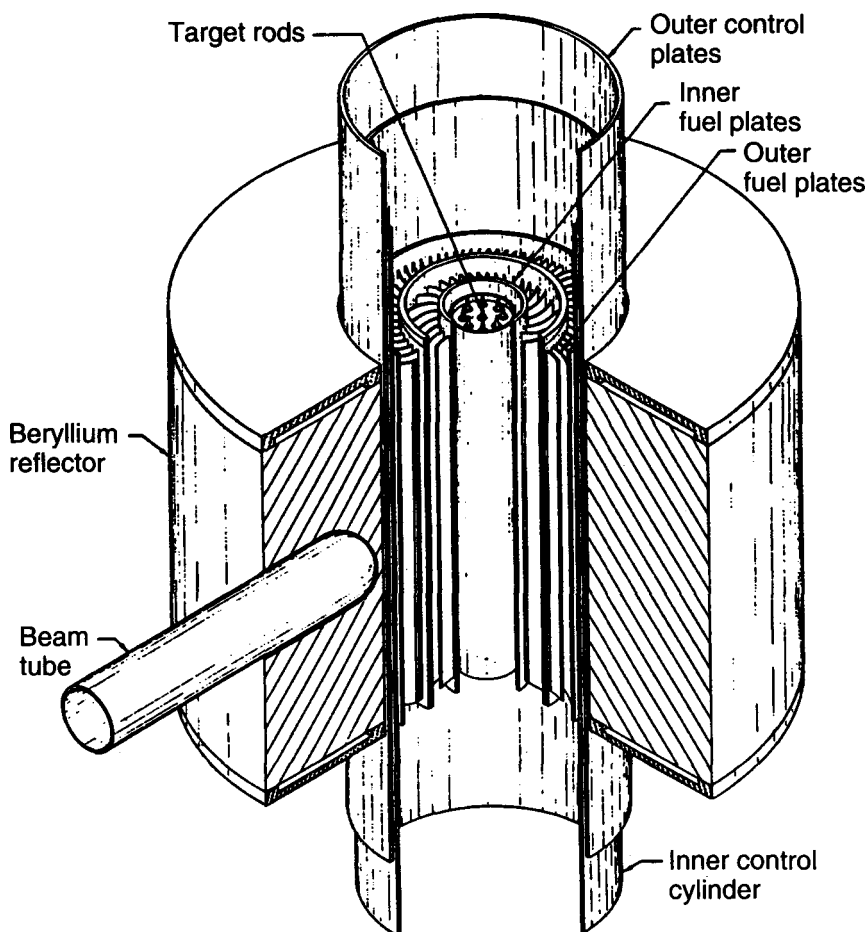


Fig. 1b Illustration of HFIR core.

analyzing for that core-melt fraction participating in a steam explosion that is energetic enough to cause vessel and/or top head bolt failure and the generation of a missile with the capacity of breaching the confinement.

SEQUENCE OF EVENTS

The scenario under consideration involves a flow-blockage-induced steam explosion phenomenon and resulting consequences. Flow blockages in the HFIR may arise from a number of foreign objects, such as badges, clear plastic wrappings, or reactor system components, that may have broken loose.

A core flow blockage of varying sizes is assumed to cause flow blockage to one or more coolant channels. The resulting flow starvation causes a sharp reduction in the heat transfer and leads to fuel-plate melting. It is then important to evaluate whether this fuel-melting

phenomenon on a localized basis will propagate to neighboring channels and to what extent. The extent of damage propagation determines the amount of core material that can participate in a steam explosion event. Briefly, steam explosions³ are physical phenomena that result from an extremely rapid thermal energy transfer between two intimately mixed liquids at different temperatures. The rapid energy transfer produces explosive vaporization rates that generate pressures and shock waves characteristic of an explosion. Various stages of steam explosions are (1) fuel-coolant mixing, (2) triggering, and (3) explosion propagation and expansion. During fuel-coolant mixing, the molten fuel gets intermixed with coolant to provide enough surface area for potential high-energy transfer rates. During the triggering phase, the fuel and coolant are brought into liquid-liquid contact whereby rapid heat transfer begins. Triggers can be spontaneous or from external stimuli. Upon triggering, the explosion propagates throughout the mixture and causes

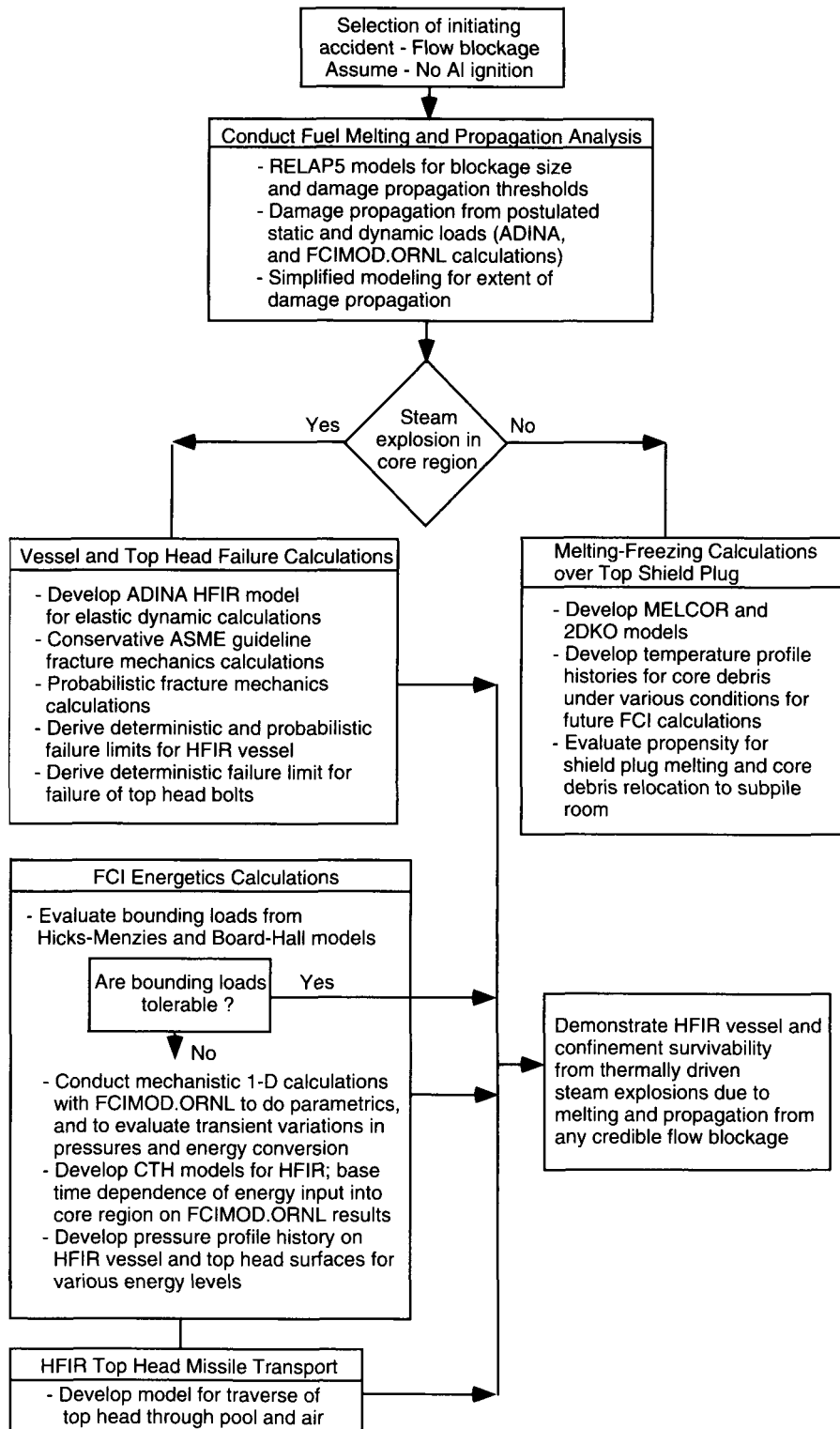


Fig. 2 Fuel-coolant interaction work elements and framework for High-Flux Isotope Reactor (HFIR) Safety Analysis Report (SAR).

high-pressure vapor formation, which performs work against the surroundings. The explosion propagation phase may or may not be accompanied by the generation of additional energy (arising from aluminum–water chemical interactions). The work done by the expanding high-pressure vapor may, in some circumstances, be sufficient to compromise the integrity of the immediate structural boundary and cause energetic missiles to form. These missiles may then penetrate confinement boundaries, allow radionuclides to bypass the filter banks, and freely enter the environment.

Because the thin (1.27-mm-thick) HFIR fuel plates of high thermal conductivity are evenly intermixed with the coolant, it is assumed that no additional premixing is necessary for a steam explosion to occur in the core region. Further, it is conservatively assumed for this analysis that a suitable triggering source will be available in the system (e.g., collapsing vapor bubbles) to permit a steam explosion to occur and propagate [i.e., we assume that if fuel melting of a given magnitude occurs, a steam explosion will occur and all the melted core will participate in the fuel–coolant interaction event (FCI)]. Note that HFIR fuel can react exothermically with molten aluminum and thereafter form a eutectic mixture with a higher viscosity. Therefore triggerability characteristics of molten HFIR fuel could be different (i.e., improved) compared with those of molten aluminum alone. On the basis of past experiences⁴ with uranium–aluminum–fueled reactors undergoing fuel-melting accidents from flow blockages, aluminum ignition simultaneous with steam explosions is assumed to be a very unlikely event. The analysis presented in this article is conducted without consideration of this additional energy source.

With this introduction, the sequence of events following a core flow blockage of a given magnitude consists of first evaluating whether a given size blockage will lead to fuel melting. If melting is predicted, several scenarios are postulated to see whether the core melting would propagate and to what extent under full power conditions. Melt propagation could occur because of static or dynamic loads. Three scenarios were postulated and analyzed. The first scenario postulates that fuel-plate melting and ablation would lead to an increase of the hydraulic diameter and flow area in affected flow channels. With the parallel channel condition, this would lead to an increase in the channel flow velocity. Beyond a certain critical velocity, the fuel plates would buckle and collapse and thus lead to core-melt propagation. The next scenario considers the situation wherein a blocked flow channel experiences flow starvation and thereby results in a circumferential pressure gradient across the adjacent fuel plates. A

sufficiently large pressure gradient may cause a large enough deflection from the static load to cause flow starvation in adjacent channels and therefore cause damage propagation. The third scenario concerns possible fuel-plate failure from dynamic pressure pulse loadings, which result from localized steam explosions. For damage propagation to occur, the pressure pulse from steam explosions should be in a position to cause sufficient plate deformation or rupture to cause melting of adjacent plates and therefore lead to the possibility of propagating steam explosions. If melt propagation is predicted, the extent of melt propagation is determined via conservative modeling, coupled with insights from past modeling conducted for the previous HFIR accident analysis.⁵ It is assumed that fuel melting will cease once the reactor is scrammed. Fuel melting and subsequent explosions may also occur under decay heating conditions if coolable geometry is lost. This determination requires a mechanistic melt progression capability, which has not yet been developed. Therefore the assumption of melt propagation arrest is predicated on the availability of a coolable geometry under postscram decay heating conditions.

The sequence of events following a determination of the extent of fuel melting from flow blockages consists of determining the energetics of resulting steam explosions coupled with interactions of loads with structural boundaries. High enough loads may cause failure of the HFIR vessel and/or top head bolts. If top head failure is predicted, a missile would form with a given initial velocity that has to travel through the large reactor pool before rising into the high bay area and possibly penetrating the confinement roof.

PROBLEM FORMULATION AND MATHEMATICAL MODELING

The specific aspects dealing with problem formulation, mathematical modeling, and computer code simulation for the various phases of steam explosion analyses are too numerous to describe here individually. Details are given in Ref. 6. Salient aspects are summarized in this section.

The approach used in evaluating the amount of core-melt fraction and fuel temperature during flow blockage events is to combine previous analyses⁵ with scoping studies, which used hand calculations and codes such as RELAP5,⁷ MELCOR,⁸ 2DKO,⁹ FCIMOD.ORN,^{10,11} and ADINA.¹² Models of various levels of sophistication were set up to determine what amount of coolant channel

area would need to be blocked before fuel-plate melting initiates and thereafter to propagate to other fuel plates. As a cross-check for RELAP5 evaluations, a simple hand calculation was first conducted on the basis of the postulate that a critical-sized flow blockage would lead to that critical mass flow rate at which liquid entering a blocked channel would be completely evaporated by the time it reaches the core exit. This calculation was constrained by the need to also maintain the pressure drop at the same level as that for unblocked channels, which would thereby satisfy the parallel channel condition. The evaluation results are given in the next section. Figure 3 shows a more sophisticated RELAP5-based model of the HFIR core set up to evaluate both the critical blockage size and multiple flow channels. The model represents ten different coolant flow channels connected between two plena.

Heat structures representing fuel plates were represented. Power profiles, nuclear feedback, hot spots, or streaks were not represented. This RELAP5 model is also capable of evaluating the effects of complete or partial flow blockage effects at core entrance. It was further extended via suitable modification to help evaluate what fraction of core plate melting in the inner or outer fuel elements would lead to core-melt propagation.

Several ADINA code models of the HFIR fuel plates were developed and coupled with imposed thermal-hydraulic boundary condition to evaluate fluid-structure interaction-induced fuel-plate failure for the three scenarios. A formulation was also set up to evaluate the critical flow velocity in enlarged flow channels caused by ablation of fuel plates. For scenario 3 events described earlier, an FCIMOD.ORNL model was set up to evaluate

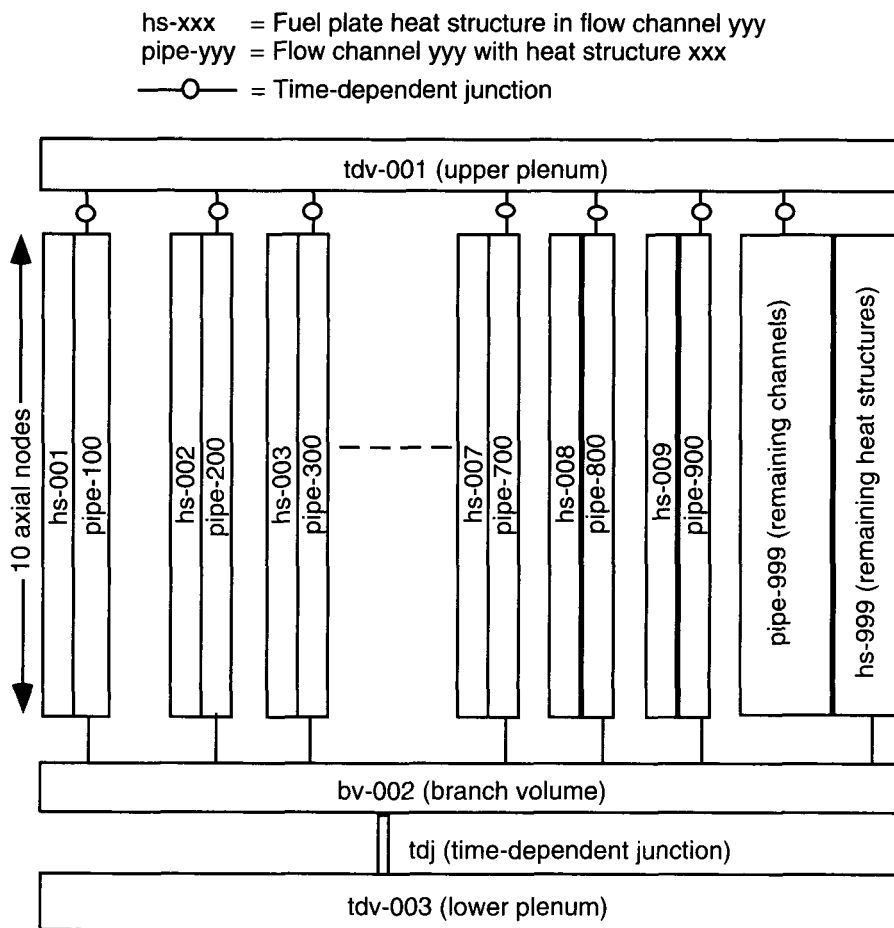


Fig. 3 Node map of High-Flux Isotope Reactor (HFIR) fuel-coolant interaction (FCI) flow blockage RELAP5 model.

pressure pulse transient behavior resulting from localized fuel melting in the HFIR core. Single and multiple fuel-plate regions were simulated. This model was coupled with ADINA models for sections of the HFIR core to evaluate failure characteristics from dynamic loads. A simple model was postulated to evaluate the degree of melt propagation before arrest via scram. The model postulates that a melted fuel plate would collapse uniformly onto its neighbor such that the plates then start to heat up and melt and collapse further in a domino fashion. The time limit available for this domino effect to keep propagating is set at 3 seconds, which corresponds to the transit time for a fluid particle to reach regions where high radiation level detection can take place, which then initiates a scram signal. Previous conservative HFIR analyses conducted and reported in Ref. 4 for small and large flow blockages are based on reactivity considerations in conjunction with the HFIR control system design. In addition to the modeling for fuel melting in the core region, 2DKO models were also set up to evaluate degrees of fuel-melt superheat possible if the core material relocates to the surface of the upper shield plug. The model was used to evaluate melting-freezing aspects for a variety of parametric studies.

A comprehensive approach was used to evaluate steam explosion energetics. Thermodynamic models were set up to evaluate maximum possible (i.e., bounding) pressure pulse magnitudes and thermal-to-mechanical energy conversion ratios. These models were based on theories of Hicks-Menzies¹³ and Board-Hall¹⁴ as implemented in the UWHM¹⁵ and UWHUGO¹⁶ codes. Bounding values of pressure rise were found to be intolerable from the standpoint of qualifying the containment potential of pressure boundaries. Hence steam explosion energetics modeling was also done with the one-dimensional (1-D) mechanistic model introduced into the FCIMOD.ORNL code and the multi-material, multidimensional shock-wave physics code CTH.¹⁷ A typical FCIMOD.ORNL model for 1-D energetics is shown in Fig. 4, whereas the best-estimate two-dimensional (2-D) HFIR model developed with CTH is shown in Fig. 5. Several additional HFIR studies were also conducted for evaluation of effects, such as the effect of the reflector, grid size, shroud, and reflecting vs. absorbing boundary conditions. Details are given in Ref. 5. The FCIMOD.ORNL models breakup and dispersion of fuel melt in a time-dependent fashion, along with heat-transfer effects during steam explosions, provide pressure-to-time and thermal-to-mechanical energy conversion histories in the explosion zone. The model was used to perform parametric studies and to provide rate dependencies for the

explosive thermal energy insertion rate in the core region for CTH calculations. As shown in Fig. 5, all the major components in the HFIR vessel have been represented, including the core region, reflector, shroud, and vessel head and walls, as well as the ability to allow for energy dissipation into the large HFIR pool and phase-change effects. Three-dimensional (3-D) effects were considered impractical to model with CTH.

A detailed modeling effort was undertaken to evaluate vessel failure^{6,18} both from a conservative deterministic sense and from a probabilistic standpoint. Both modeling approaches used the principles of fracture mechanics. An ADINA code model of the HFIR vessel was developed to

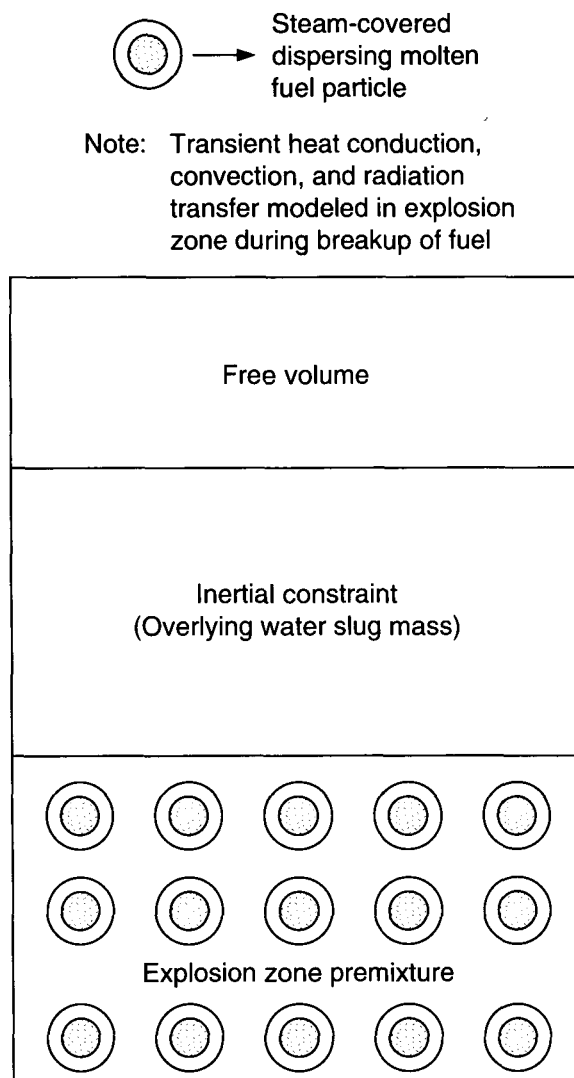


Fig. 4 FCIMOD.ORNL modeling geometry.

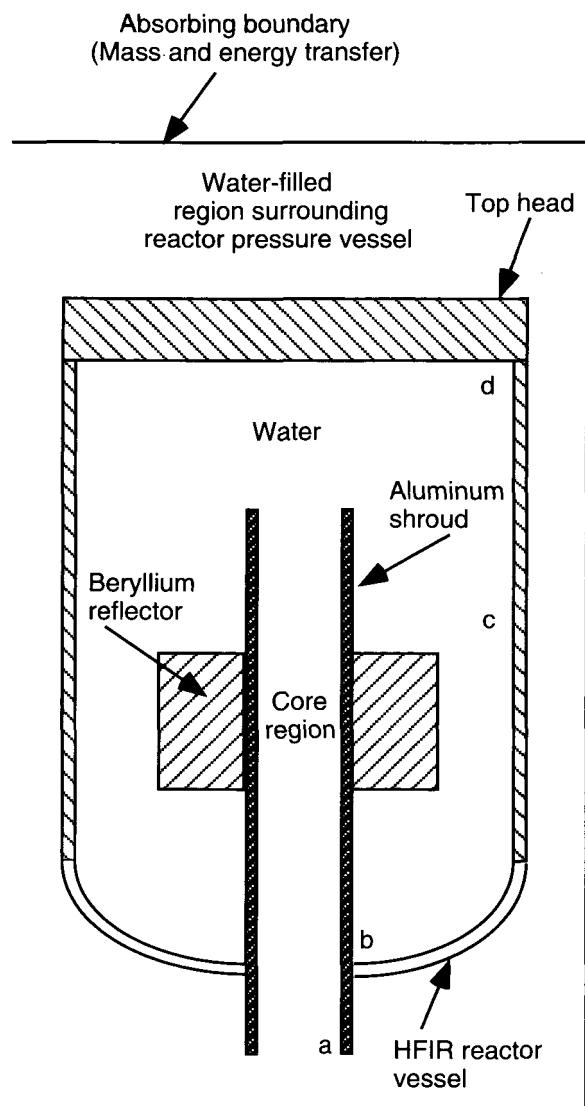


Fig. 5 CTH model for two-dimensional simulation of steam explosion dynamics.

evaluate critical stresses for failure. Dynamic pressure pulses of a given magnitude and duration are used in conjunction with conservative American Society of Mechanical Engineers (ASME) guidelines for specification of cracks and membrane response to evaluate a so-called fracture toughness and geometric factor. From this is an allowable hoop stress above which failure is predicted and evaluated. Cheverton's data base,¹⁹ in conjunction with dynamic stresses predicted from the ADINA HFIR

pressure vessel model, was used to develop a second modeling approach to calculate fracture probability of the vessel.

For the evaluation of top head bolt failure, the HFIR top head was represented as an equivalent circular disk with the assumption that the 44 bolts securing the head to the vessel uniformly absorb steam explosion loads. Thereafter, for a given dynamic pressure imposed on the lower surface of the disk, the effective stresses in the bolts are calculated. A failure criterion was developed that postulated that bolt failure would occur if the effective stress in the bolts exceeds the material yield stress for more than 0.6 ms (a time span taken from analysis of failure curves developed for the HFIR vessel as described later).

The model formulation for top head missile transport through the HFIR pool consisted of setting up and solving a pair of differential equations describing the motion of a disk through water. The model takes into account viscous drag, inertia, gravitational deceleration, and virtual mass forces (to model fluid displacement ahead of moving disk). Modeling of plume formation was not considered important for the HFIR SAR evaluations.

CONSEQUENCE ANALYSIS

The modeling framework described earlier was used to conduct a comprehensive analysis of steam explosion events arising from core flow blockages in HFIR. Because of the large volume of information generated, only highlights of the analysis results are provided in this section. Reference 5 should be consulted for more information.

Analysis of Flow-Blockage-Induced Fuel Melting and Propagation

The RELAP5 model of Fig. 2 coupled with hand calculations was first used to analyze whether melting would occur if a single flow channel were completely blocked. We found that a complete single-channel flow blockage will not lead to fuel melting. Sufficient heat-transfer capability exists in the unblocked side of the fuel plate to convect the fission heat generated. However, as may be expected, complete flow blockage of two adjacent flow channels to a fuel plate does lead to fuel melting. The simple model for evaluating the critical blocked area corresponding to a critical mass flow rate predicted that about 74% of multiple blocked flow channels would be necessary to cause fuel melting in the blocked channels. Again, for multiple blocked flow channels, the more sophisticated RELAP5 model of Fig. 3 revealed that

blockage of several flow channels above 92% of the collective flow area would be necessary to cause the onset of fuel-plate melting. Figure 6 shows a variation of core exit void fraction and fuel-plate temperature for the 92% flow blockage case. Exit void fraction rises sharply to about 60% at the onset of flow instability before reaching Critical Heat Flux (CHF) conditions. Upon CHF occurrence, fuel-plate temperature rises rapidly (almost adiabatically) to the aluminum melting temperature of 660 °C, after which it is artificially held constant. RELAP5 modeling does not capture multidimensional phenomena, such as effects of stagnation zones downstream of obstacles and material swelling. Such effects are the subject of future studies. Nevertheless, these conservatively scoped 1-D calculations do indicate that a substantial portion of the inlet to several flow channels would need to be blocked before fuel melting can ensue. Valuable information regarding initial thermal-hydraulic conditions at the onset of steam explosions was also derived.

The modified RELAP5 model was next used to analyze damage propagation characteristics for inner and outer element fuel plates. The analysis revealed that up to nine or seven plates in the inner or outer fuel-element regions could melt without causing the neighboring fuel plates to heat up to melting conditions.

Separate calculations conducted with the ADINA code models for fuel plates revealed that fuel-plate ablation would not lead to coolant velocities large enough to cause buckling instabilities (viz., from scenario 1). The same models also revealed that excessive fuel-plate deflections would not result from circumferentially imposed static loads (viz., from scenario 2). Hence damage propagation from these postulated scenarios is highly unlikely during flow blockage events. The ADINA models for inner and outer HFIR fuel plates under a variety of conditions were used to evaluate dynamic failure envelopes of fuel plates subjected to dynamic steam explosion loads. Briefly, ADINA model results indicated that HFIR core fuel plates would fail if subjected to triangular-shaped steam explosion pressure pulses of magnitude greater than 0.18 MPa in the millisecond duration range. These results were combined with FCIMOD.ORNL results of pressure pulse histories generated for one and two fuel-plate melting conditions as shown for selected cases in Fig. 7. Insights gained from RELAP5 results regarding initial thermal-hydraulic conditions were used in FCIMOD.ORNL steam explosion evaluations. It was determined that localized steam explosion loads would likely result in failure of adjacent fuel plates, which represents a potential mechanism for core-melt propagation.

The postulated conservative model was next used for determining the extent of damage propagation over 3 seconds from a domino effect caused by melted plates successively collapsing on neighboring fuel plates. This analysis revealed that a maximum of about 77 plates, or 14% of the core fuel, could melt under full power conditions before the process is halted via actuation of the scram function. This value is actually lower than that of the bounding model (of Ref. 4) evaluation of 24% of core mass melting from large flow blockages. This bounding model is based on the specific reactivity compensation feature of HFIR. The HFIR control system compensates for a 1 dollar reactivity change, after which a scram signal is actuated. Briefly, in this bounding model, a flow blockage over a given number of plates is assumed to lead to a loss of heat transfer for those plates coupled with a temperature rise in coolant and fuel plates. This leads to a corresponding change in core reactivity, which is tied back to that size blockage that will correspond to a reactivity change of 1 dollar. Thereafter scram occurs relatively instantaneously. Because a mechanistic capability for core-melt progression has not yet been developed, it was conservatively assumed from the flow blockage scenario that up to 24% of the HFIR core material may melt and participate in a steam explosion event "under full power conditions."

Debris heatup over the top shield plug was analyzed with the use of the 2DKO and MELCOR models. These heat-transfer calculations indicated that a potential exists for melt superheat to occur if the core debris melts and relocates onto the lower shield plug region. This potential is a function of several parameters (viz., debris power density upon release of fission products, aluminum ignition, amount of debris discharged, etc.). Because of resource constraints, this configuration was not possible to specifically analyze further to evaluate steam explosion energetics coupled with pressure boundary failure and missile generation. However, with the use of engineering judgment, it appears that the resulting pressurization loads and the generation of a confinement-damaging energetic missile may be lesser under these conditions than under steam explosion conditions in the core region (which have been analyzed extensively). This engineering judgment is predicated on having similar thermal-hydraulic conditions for the debris over the shield plug and in the core region. Again, a steam explosion of similar intensity over the shield plug region would have to overcome a significantly larger inertial water mass as well as structural material compared with steam explosions occurring in the core region.

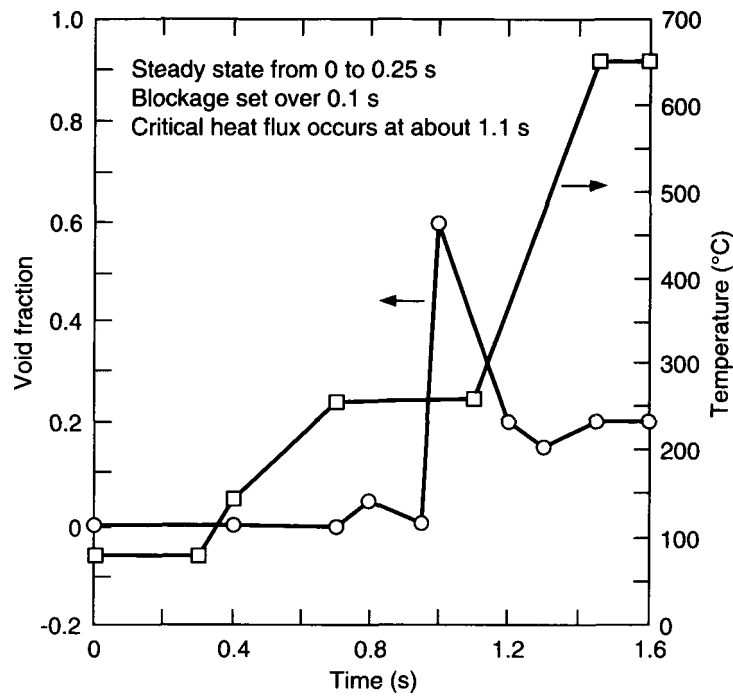


Fig. 6 Variation of core exit void fraction and plate temperature for 92% flow blockage case.

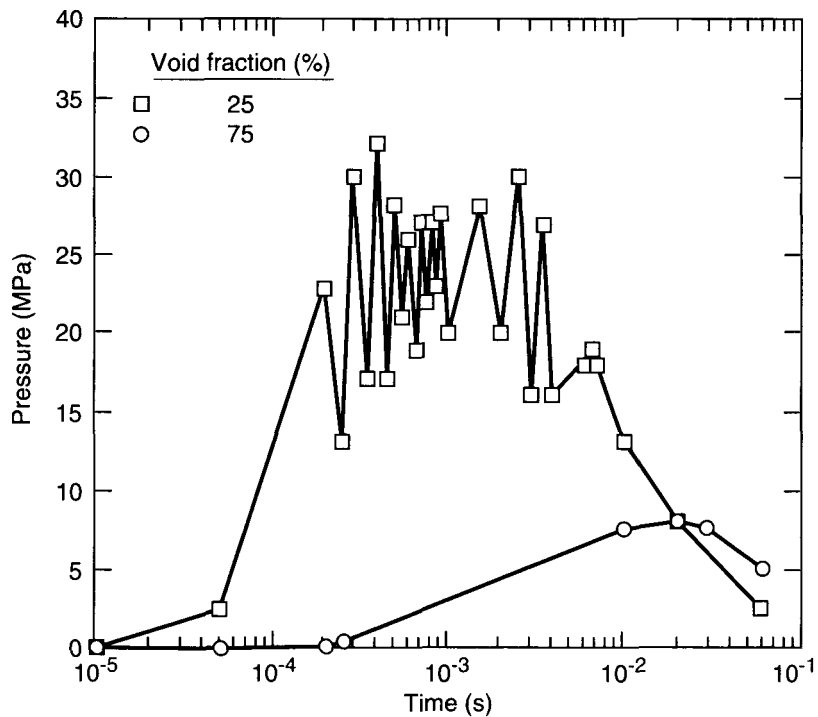


Fig. 7 Variation of pressure with time in explosion zone for two initial void fractions.

Vessel and Bolts Failure Analysis

The analysis work done for evaluating vessel and bolts failure is described separately.

Vessel Failure Characteristics

Elastic dynamic ADINA calculations were performed to obtain hoop stress magnitudes at the three locations shown in Fig. 5. The effective stress values for the points "a" and "b" are essentially similar. For point "d," however, the effective stresses in the material were significantly higher. This is to be expected because point "d" is at a location where significant stress concentrations can occur. Point "d" values are not considered here because, in reality, the top head is bolted to the vessel. Therefore, for the evaluation of vessel failure, the failure envelopes at the vessel midplane will be taken as representative. So-called failure envelopes generated for the HFIR vessel wall are shown in Figs. 8 and 9. These are essentially plots of peak-induced stresses in the vessel wall when subjected to an external pressure pulse (y-axis) of a given magnitude (x-axis). As shown in Figs. 8 and 9, the failure curves tend to flatten out for pulse durations larger than about 0.6 to 0.8 ms. For analysis purposes, this implies that, to determine whether a steam explosion will cause vessel failure, one need only determine if the pressure pulse of a given duration lasts for more than 0.6 to 0.8 ms. Therefore the precise value of pulse duration is not as important.

Thereafter a conservative deterministic estimate was made to evaluate vessel failure loads on the basis of ASME guidelines as mentioned earlier. This resulted in a failure hoop stress of about 245 MPa (35 ksi). If the information shown in Fig. 8 is combined with the knowledge of the allowable vessel effective/hoop stress of 245 MPa (35 ksi) lasting for more than about 0.6 ms, we note that the largest pressure pulse that can be tolerated is no more than 10 MPa (1.5 ksi). This approach gives rise to very conservative estimates for vessel failure loads because it uses a set of highly conservative ASME guidelines. Therefore efforts were put in place to analyze HFIR vessel integrity from a best-estimate probabilistic view. Results from this study indicated that the probability of fracture is to the order of 10^{-5} after 10 effective full-power years (EFPY) of embrittlement since 1986 for the material stress level of 161 MPa (23 ksi). The probability increases to the order of 10^{-2} as the material stress increases to 518 MPa (74 ksi) as shown in Fig. 9.

On the basis of the probabilistic approach results, it was concluded that the 10-MPa failure pressure [corresponding to a hoop stress of 245 MPa (i.e., 35 ksi)] has a

low fracture probability of 10^{-4} . For higher values of hoop stresses [viz., >500 MPa (or >70 ksi)], however, the corresponding failure pressure rises to about 21 MPa with a higher fracture probability (approaching 10^{-2}).

Top Head Bolt Failure Analysis

On the basis of the model for top head failure described previously, an analysis was conducted to evaluate what level of loads would be sufficient to cause the bolts (holding the top head to the vessel) to fail.

An important feature of the analysis for bolts failure is related to the time duration of the pressure pulse. As is well known, permissible material stress levels can increase quite sharply if the duration of the imposed pressure pulse gets smaller and smaller. Such an evaluation would require a dynamic structural analysis. Advantage was taken of the results of vessel failure analysis shown as failure curves for the HFIR pressure vessel (viz., Fig. 8) to provide guidance on the time duration of pulses necessary, after which the failure curve tends to flatten out. The ratio of average stress in the bolts to imposed hydraulic pressure (at the bottom surface of the top head) was calculated to be about 24. Original HFIR drawings give the yield strength of steel bolts to be about 840 MPa. Therefore, to exceed the yield stress in the bolts (which are pretensioned after refueling), the imposed hydraulic pressure required to break the bolts needs to be in excess of 26 MPa.

A probabilistic fracture-mechanics study for the bolts region as was done for the vessel would be necessary to judge whether the bolts would fail before the vessel. A steam explosion in the core region results in the largest loads on the top head as a result of the channeling effect caused by the shroud and reflector followed by the side walls. Even for the vessel side walls, pressure loadings are greatest at the intersection of the top head and vessel wall with the vessel midplane region being loaded to about half of that at the vessel-head intersection. If we couple this with the ADINA model results, which indicated that stress concentrations are greatest at the intersection of the top head and vessel side walls, failure should be expected either by bolts breaking or the vessel ripping around at the vessel-head intersection. For both failure modes, a top head missile would result. In the absence of a detailed study of bolts failure (similar to that done for the vessel), the conclusion can be drawn that if the pressure level adjacent to the top head lower surface exceeds the level of about 26 MPa for more than about 0.6 to 0.8 ms, the bolts would fail. For any extent of time that the imposed pressure exceeds this range, the effect

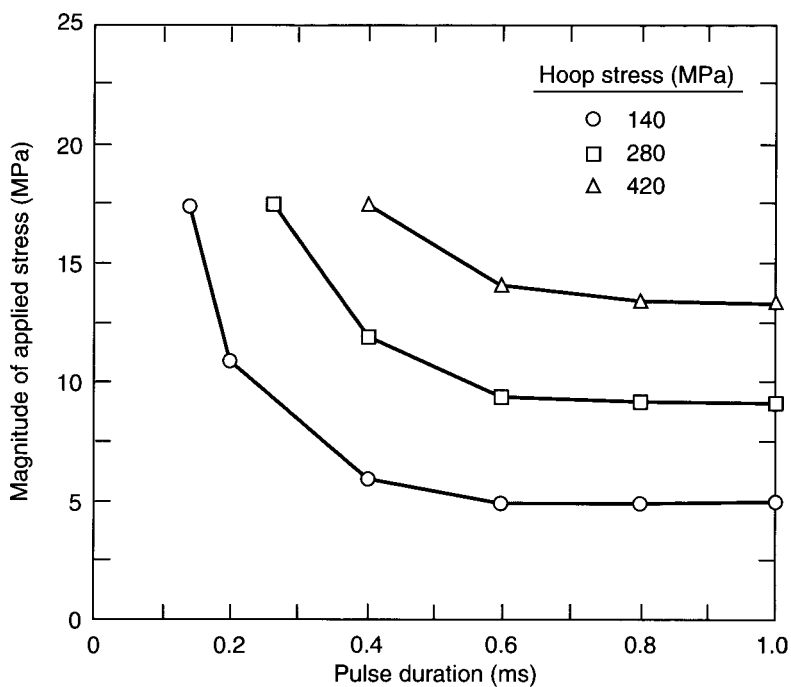


Fig. 8 Failure envelopes for High-Flux Isotope Reactor (HFIR) pressure vessel wall at midplane.

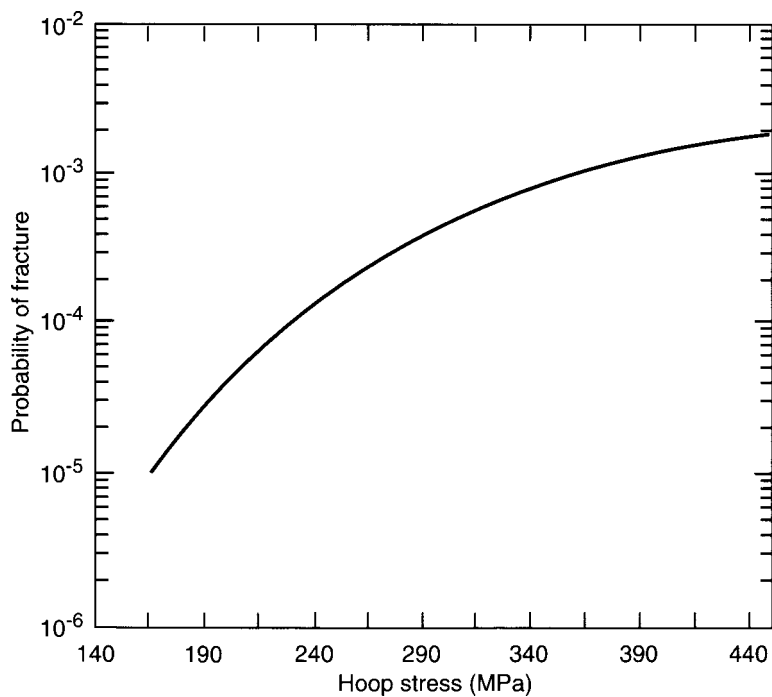


Fig. 9 Failure envelopes for variation of HFIR vessel failure probability vs. hoop stress (0.007 crack/ft^2).

would tend to be one where momentum transfer occurs to accelerate the top head. In reality, upon bolts failure and top head rise, pressure relief may also occur to the large reactor pool. Pressure relief may also occur under asymmetric loading of the vessel and top head (the determination of which would require a 3-D simulation). This would then tend to lessen the degree of momentum transferred to the top head, which indicates that the current modeling approach is conservative (the degree of which is difficult to determine). A detailed study of such a pressure relief was not possible to conduct for the work done for the HFIR SAR. Hence, from the standpoint of conservatism, the effect of pressure relief is not given credit in evaluating missile energetics.

Analysis of Steam Explosion Energetics in HFIR

Steam explosion energetics calculations for several cases were conducted first with thermodynamic models to obtain upper-bound pressurization and thermal-to-mechanical energy conversion values. This preliminary analysis indicated that, depending on the thermal-hydraulic conditions, resulting pressurization levels could range from 60 MPa to several hundred megapascals, with the conversion ratio varying from a low value to about 45%. This bounding analysis proved useful in indicating trends of important parametric variations and providing guidance for setting upper-bound limits for judging the validity of multidimensional model predictions. These values were not used to evaluate HFIR safety characteristics from steam explosion events. Significant reductions in pressurization and conversion ratio predictions were observed for the same conditions when using the FCIMOD.ORNL and CTH models. Typical FCIMOD.ORNL pressure and mechanical energy conversion source-term results for HFIR conditions are shown in Fig. 10. Figure 10 indicates that most of the thermal energy transfer causing pressure buildup is over in the first millisecond or two. Results of FCIMOD.ORNL calculations were used to generate energy source-term rate values for multidimensional CTH calculations conducted with the best-estimate model of Fig. 5. The model was exercised with energy deposition levels of 7, 31, 51, and 65 MJ, which represent core-melt fractions of approximately 7, 30, 50, and 65%, respectively. The cases with 7 and 31 MJ of thermal energy inserted in the explosion zone did not result in sustained pressure levels in excess of failure levels for the vessel or top head bolts. The case with 31 MJ of energy deposition does give pressure pulses in the centerline region right

under the top head greater than 26 MPa. However, these are peak pulse magnitudes and do not last for more than about 0.1 to 0.3 ms; therefore the impulse transferred to the top head bolts is smaller than that required for failure to occur. In addition, the pulse magnitude decreases significantly from the centerline to the vessel wall interface region, with the result that vessel failure pressure level (of 21 MPa lasting for more than 0.6 ms) is not reached. These attributes are clearly seen in the sample results displayed in Fig. 11 for the 31- and 51-MJ energy insertion cases. An important aspect of the situation for the 7- and 31-MJ cases relates to the fact that the mechanical integrity of the aluminum shroud tube is not affected. This accounts to a large measure for the significant variation in pressure pulse magnitudes from the top head centerline to the vessel wall-top head interface. For these instances, the shroud acts as a sort of channel, directing pressure waves upward, and thus limits the degree of dissipation in the radial direction. It also serves as a kind of organ pipe giving rise to significant ringing effects as seen in the high-frequency pressure waves being built up as the transient progresses and reflected waves tend to overlap. For the 7- and 31-MJ cases, significant reduction in pressure pulse levels occurs in the radial direction as a result of this organ-pipe effect.

The two additional cases with 51- and 65-MJ energy insertion did cause the aluminum shroud to rupture from the FCI energetics. The rupture of the shroud allows for increased dissipation of explosion energy in the radial direction and also leads to significant reduction or even elimination of the buildup with the preceding organ-pipe effect. Pressure pulse histories for the 51-MJ case (directly beneath the top head in line with the vessel centerline and also in the explosion zone) are shown in Fig. 11. The pressure pulse magnitudes underneath the top head display much less variation in the radial direction than in the earlier cases where the shroud had not ruptured. For the 51- and 65-MJ cases, the average pressure below the top head and in the vicinity of the reactor vessel is larger than the required 21-MPa pressure (lasting more than 0.6 ms) required for vessel rupture from fracture, or even the 26 MPa required for failure of bolts and thereafter for generation of an energetic missile. These results would indicate that the energy level required to cause imminent vessel failure would amount to a value between 31 and 51 MJ. Engineering judgment indicates that this value is likely around the 40-MJ energy level, which conservatively corresponds to a core-melt fraction of about 40% if there are no aluminum-water chemical interactions. For the 51-MJ case, the average pressure over the top head lower surface amounts to

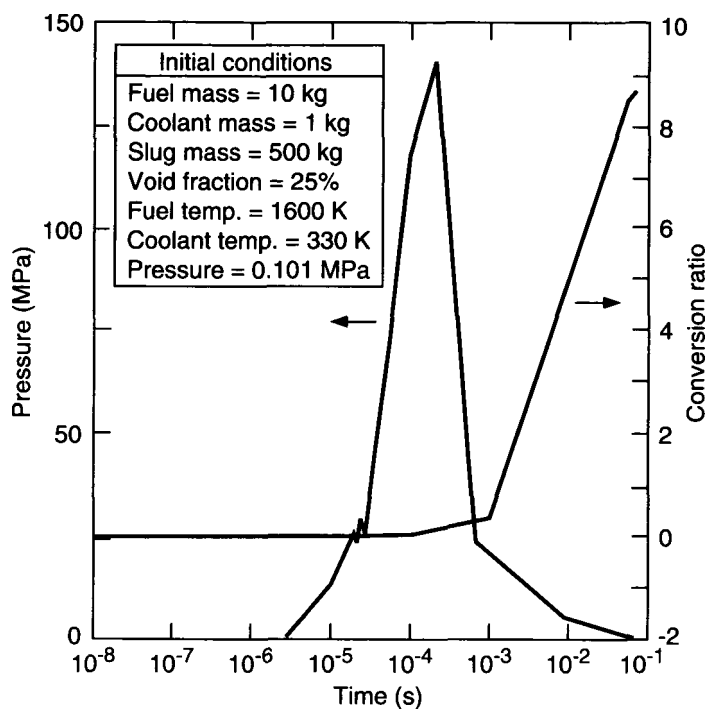


Fig. 10 Typical variation of pressure and conversion ratio with time.

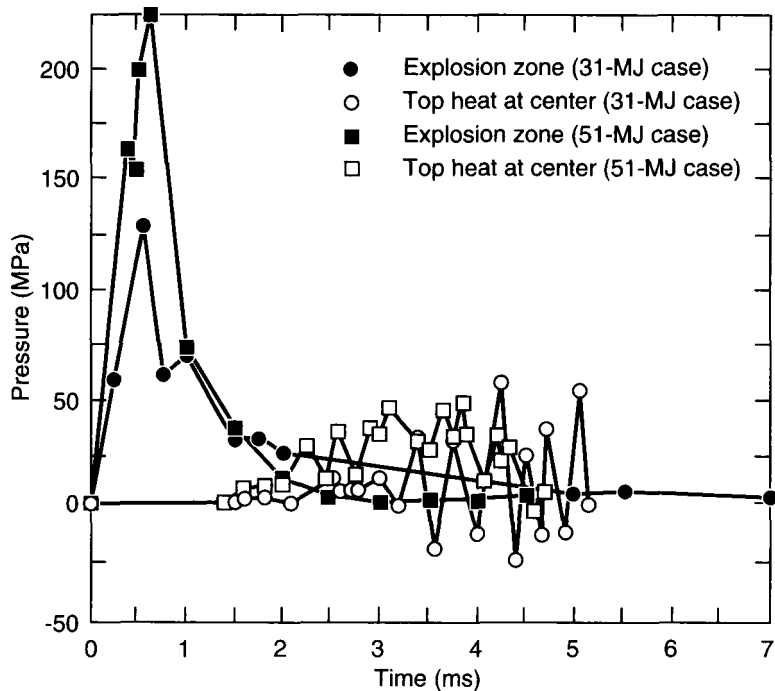


Fig. 11 Variation of pressure in explosion zone and top head center lower surface for 31- and 51-MJ energy deposition cases.

about 30 MPa and lasts about 3 ms, whereas the corresponding values for the 65 MJ case are in the vicinity of about 35 MPa and also last around 3 ms. Further details are given in Ref. 5. Another important result worth noting is the sharp reduction in pressure levels from the explosion zone to the top head and vessel wall boundaries. As shown in Fig. 11, pressures in the explosion zone can be higher by a factor of 5 or larger than pressures at the system boundary (for the HFIR case). This clearly attests to the importance of including multidimensional effects.

For the cases where vessel or bolts failure may occur, it is necessary to evaluate what, if any, the initial velocity of a missile might be. If we estimate that an average pressure (P_{av}) acts on the top head for a given time (t) after the top head has broken loose, the initial upward velocity of the top head is estimated from a momentum balance formulation. In this formulation, it is conservatively assumed that no pressure relief occurs to the reactor pool when the bolts fail, and the top head lifts off as a missile. The analysis for the 51- and 65-MJ cases revealed that P_{av} values of about 30 and 37 MPa are experienced over the top head lower surface for t values of approximately 2.4 to 3 ms, respectively. This gives rise to initial velocity of about 25 to 37 m/s for the 51-MJ case and an initial velocity of about 30 to 37 m/s for the 65-MJ case.

These estimated velocities were used to evaluate missile energetics and transport through the reactor pool and high bay air space.

Missile Evolution and Transport Analysis

An analysis was conducted to evaluate top head missile energetics for situations where a steam explosion of sufficient intensity causes the bolts to break and accelerate the top head (~14 000 kg mass) with a prescribed initial velocity. The model formulation of this phenomenon described in Ref. 5 was used to evaluate top head transport characteristics through the 4.3-m(14-ft)-deep reactor pool filled with water. Results were obtained for a conservative and best-estimate value for the drag coefficient, C_d . For $C_d = 1.0$ (i.e., conservative value) and for initial velocities of 20 and 35 m/s, the rise height above the pool surface amounts to 3.5 and 14.6 m, respectively. On the other hand, with $C_d = 1.2$ (i.e., best-estimate value), the corresponding rise heights are calculated to be 1.6 and 8.2 m, respectively.

As mentioned previously, for the initial velocity in a case where about 65-MJ energy level is inserted in the core region, the top head initial velocity would be in the vicinity of 30 to 37 m/s. As shown from the preceding

calculations, such an initial velocity may be considered a threshold velocity for causing the top head to almost reach the confinement roof, which is about 14 m (48 ft) above the pool surface level⁴ if the drag coefficient were 1.0.

CONCLUDING REMARKS

On the basis of the results presented in this article and using engineering judgment to conservatively account for uncertainties, we thus conclude that, to threaten the HFIR confinement and cause bypass of filter banks, about 65 MJ of thermal energy would need to be inserted into the reactor core region on an explosive time scale. This conservatively amounts to about 65% core-melt participation in a steam explosion event. As mentioned previously, the underlying assumption is that aluminum temperatures will not rise to high enough levels to cause ignition.

On the basis of the front-end work done to evaluate the level of core melting from flow blockage events, it is considered highly unlikely that 65% or more of the HFIR core can melt and materially participate in a steam explosion event. From the analysis results presented in this article, it has been shown that the maximum possible core-melt fraction would range from about 14% to about 24%. These levels of core melting are not even high enough to cause vessel or top head bolts failure, which, as demonstrated earlier, requires about 40% of core-melt participation in a steam explosion event. As mentioned earlier, a key assumption made in the HFIR FCI analysis during flow blockage events is that aluminum ignition will not occur. This assumption gains some credibility from past experiences with uranium-aluminum-fueled reactors undergoing fuel melting accidents from flow blockages where aluminum ignition did not occur. In general, this remains an open issue, the determination of which (for HFIR conditions) would require an adequate core-melt progression study to give appropriate estimates of initial conditions (viz., amount of melting and degree of superheat). This aspect is currently under research for many DOE reactors and also for the HFIR. It is expected that these studies, coupled with the unique nature of molten HFIR fuel, will demonstrate the unlikely nature of chemical reactions occurring on an explosive time scale in HFIR (during flow blockage events).

On the basis of the available evidence, it is judged that the HFIR pressure vessel and top head structure will be able to withstand loads generated from thermally driven (i.e., no ignition) steam explosions initiated by any credible flow blockage.

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An Analysis of Disassembling the Radial Reflector of a Thermionic Space Nuclear Reactor Power System

By M. S. El-Genk and D. V. Paramonov^a

Abstract: *An analysis was performed to investigate the effect of disassembling the radial reflector of the TOPAZ-II space nuclear reactor following a postulated reactivity initiated accident (RIA). In this RIA, the control drums, starting in the full-in position, are assumed to run out at their maximum speed of 1.4°/s to their full-out position and remain out. This noncredible event occurs because of a malfunction in the drive mechanism of the control drums. Results indicate that the disassembly of only 3 of 12 radial reflector panels would successfully shut down the reactor with little overheating of the fuel and the moderator.*

The Russian TOPAZ-II space nuclear reactor thermionic power system is designed to produce up to 6 kW of electricity for at least 3 years. To "leapfrog" the system level experience and capitalize on the Russians' experience with thermionic (TI) systems, the U.S. Government purchased a number of the Russian TOPAZ-II system units with electrically heated thermionic fuel elements (TFEs) for testing at the Thermionic System Evaluation Test (TSET) facility in Albuquerque, N. Mex.¹ The knowledge gained from TSET will be incorporated into the ongoing effort by industry to develop thermionic space nuclear reactor power systems.

Extensive system analyses are currently being performed by the Air Force Phillips Laboratory (AFPL) and other members of the New Mexico Thermionic Alliance (namely, Los Alamos National Laboratory, Sandia National Laboratories, and the University of New Mexico) to investigate the safety and operation characteristics of the TOPAZ-II system during both steady-state and transient operations.

For the proposed Nuclear Electric Propulsion Space Test Program (NEPSTP), a TOPAZ-II reactor will be used to power electric propulsion devices. For the very high initial orbit (5250 km), electric propulsion devices

will be used to increase orbital altitude while conducting scientific measurements. At this high altitude, operational accidents should have no significant effect on the earth and its population.² Nonetheless, it is useful to explore noncredible events to bound the consequences and to provide information for a probabilistic risk assessment.

The objective of this article is to assess the effect of the disassembly of the radial reflector of the TOPAZ-II reactor as well as to determine the minimum number of the radial reflector panels that need to be disassembled to shut down the TOPAZ-II reactor following a postulated reactivity initiated accident (RIA). In this RIA, the control drums, starting in the full-in position, are assumed to run out at their maximum speed of 1.4°/s to their full-out position and remain out. The Thermionic Transient Analysis Model (TITAM)³⁻⁹ is used to explore this noncredible accident, which is assumed to occur because of a malfunction in the drive mechanism of the control drums. In addition to the temperatures of the different core components (fuel, moderator, coolant, core support plates, and TFE electrodes), the reactivity excursion and feedback effects in the reactor core are calculated, before and after the disassembly of the radial reflector panels, as functions of time during the transient.

SYSTEM DESCRIPTION

The primary components of the TOPAZ-II space nuclear reactor power system are (1) sodium potassium (NaK) (78%) cooled nuclear reactor with an epithermal neutron energy spectrum, (2) electromagnetic (EM) pump for circulating the coolant through the reactor coolant loop and the radiator, (3) lithium hydride radiation shadow shield, (4) volume accumulator, and (5) radiator for heat rejection into space. Other important components include startup batteries for the EM pump; cesium reservoir assembly, helium gas system, and instrumentation and control subsystem. A schematic of the TOPAZ-II space nuclear power system is shown in Fig. 1.

^aThe University of New Mexico, Institute for Space Nuclear Power Studies, Albuquerque, N. Mex.

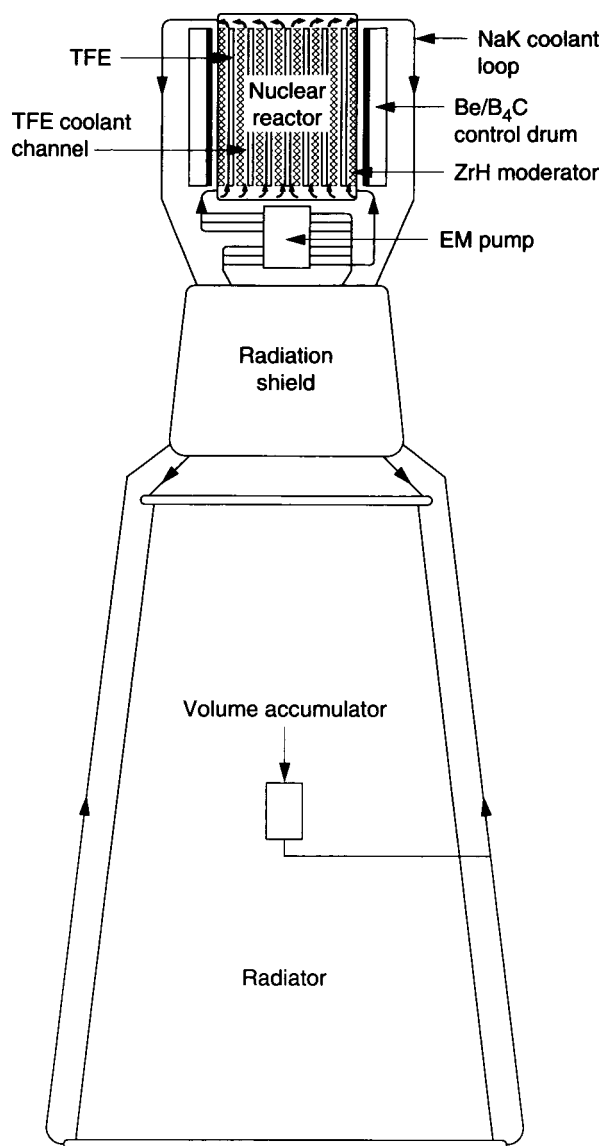


Fig. 1 Schematic of the TOPAZ-II Space Nuclear Reactor Power System. TFE is thermionic fuel element; EM is electromagnetic.

The reactor core is a right circular cylinder with monolithic zirconium hydride ($\text{ZrH}_{1.8}$) moderator blocks. These blocks are covered with a CO_2 -He-based gas mixture and coated with a special sealer to minimize hydrogen losses during reactor operation. The moderator blocks are contained in a stainless steel canister with 37 circular vertical channels that are arranged in a triangular lattice. Each channel accommodates a TFE and its coolant duct. Of the 37 TFEs in the TOPAZ-II reactor, 3 are connected electrically in parallel to supply power to the EM pump and the other 34 TFEs are connected in series

to supply up to 6 kW of electric power to the load at 27 ± 0.8 V dc. During nominal operation at 115 kW thermal power, the EM pump consumes 750 A at about 0.35 V while maintaining a total coolant flow rate of approximately 1.3 kg/s.

The coolant for the TFEs flows through annular channels between the stainless steel cladding and the moderator canister wall. The reactor is fueled with highly enriched UO_2 pellets, with a central hole for venting fission gases, stacked inside the cylindrical emitter tubes of the TFEs. The thin-walled, stainless steel vessel of the core supports the TFEs and provides plena for the NaK coolant, the helium gas for the TFEs sheath/insulator gap, and the cesium vapor.¹⁰

A detailed description of the TOPAZ-II nuclear reactor system, the design parameters, and dimensions of the TFEs is available elsewhere.⁷⁻¹⁰

Figure 2 contains a radial cross-sectional view of the TOPAZ-II nuclear reactor core showing the arrangements of the TFEs in the core and of the safety and control drums in the radial reflector. In addition to the axial beryllium (Be) and beryllia (BeO) reflector at the bottom and the top of the reactor core, respectively, the stainless steel vessel of the reactor is surrounded by a radial Be reflector with 12 $\text{Be/B}_4\text{C}$ rotating safety and control drums. These drums are divided into two groups: safety and control. The first group consists of three safety drums with a total reactivity worth of 2 dollars and a single rotation speed of 22.5°/s. The second group is comprised of nine control drums with a total reactivity worth of 4 dollars and 80 cents and can be operated at angular speeds up to 1.4°/s.

The radial reflector, including both the safety and control drums, is held together by retention metal straps that can be served by command or during reentry heating. The assembly of these straps, which measure 10 mm by 0.5 mm in cross section, is similar to that of the SNAP-10A. They are kept closed with two electric locks with melt-able stainless steel elements. In case of an emergency, the stainless steel elements are melted on command by passing an electric current through them or by reentry heating, which unlocks the metal straps. Subsequently the radial reflector is disassembled with the aid of compression springs. Unlocking the retention metal straps and disassembling the reflector take less than 0.5 second.

MODEL DESCRIPTION

A version of the TITAM has been developed for the TOPAZ-II space nuclear reactor power system by

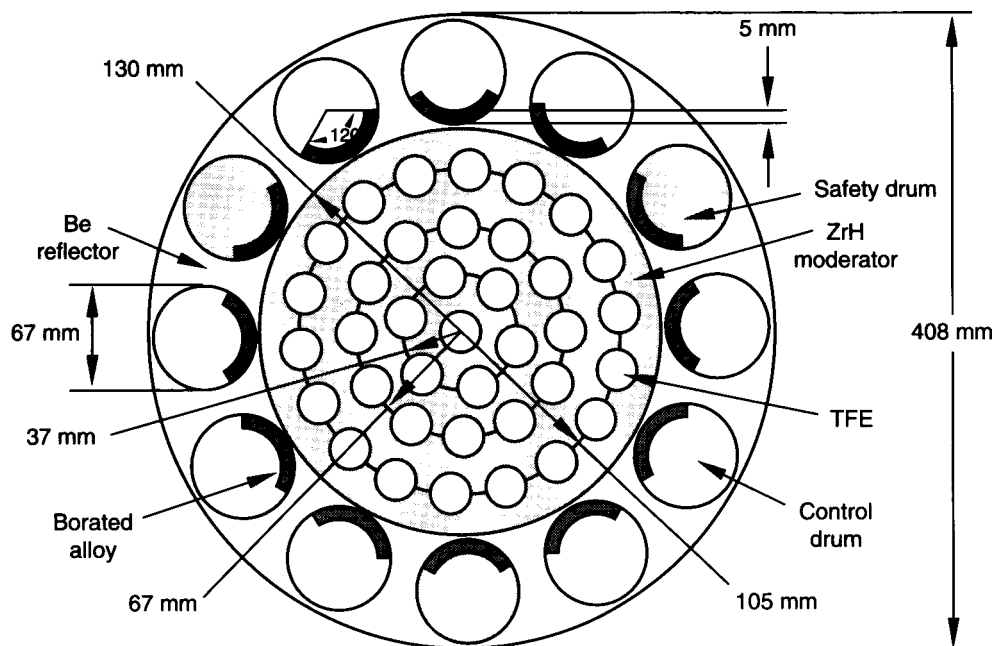


Fig. 2 Radial cross-sectional view of the TOPAZ-II reactor core. TFE is thermionic fuel element.

thermal-hydraulic coupling of the reactor model in TITAM to the power system primary loop and radiator.⁷⁻⁹ Figure 3 shows a line diagram of the TITAM model for the TOPAZ-II system, which consists of a reactor model, a coolant loop thermal-hydraulic model, an EM pump model, a radiator model, and a volume-accumulator model. The thermal-hydraulic model couples these component submodels through the system's overall energy

and momentum balance equations. The TOPAZ-II reactor model in TITAM is based on a single TFE that is thermally coupled to an equivalent cell of the zirconium hydride moderator having an adiabatic outer surface.⁷⁻⁹ The reactor model consists of several intercoupled submodels: (1) a six-group point-kinetics model; (2) a one-dimensional transient thermal model of a fully integrated, single-cell TFE; (3) an electric circuit model for

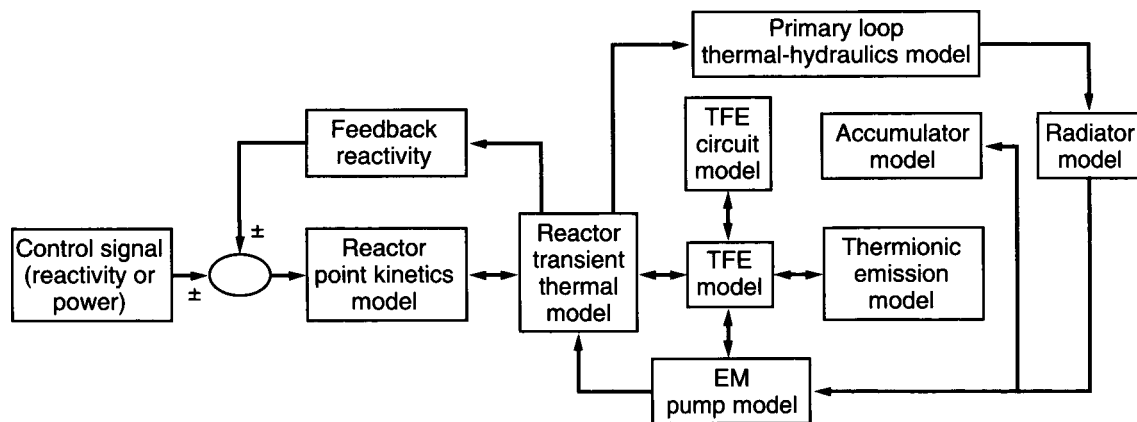


Fig. 3 Line diagram of TITAM for the TOPAZ-II Space Nuclear Power System. TFE is thermionic fuel element; EM is electromagnetic.

the TFEs; and (4) a thermionic-emission model.¹¹ More details on the description and verification of these models are available in Refs. 2 to 9.

The heat losses from the coolant loop structure by radiation to space is assumed to be 3.5% of the thermal energy removed from the reactor core. Thermal end losses of the electrodes are accounted for in the TFE model. The effective radiator area for TOPAZ-II is 7.2 m², and the mass of the radiator and of the primary loop structure is 50 kg each. The structure material of the primary loop is stainless steel, whereas that of the radiator is 80% stainless steel for piping and 20% copper for radiation fins. Fission heating of the TFEs is assumed uniform along their length, and about 4% of the fission power is deposited in the moderator. The fill gas in the gap between the stainless steel canister and the ZrH moderator blocks is taken to be CO₂. The thermophysical properties of the electrode materials, He and CO₂ gases, Cs, stainless steel, coolant, and moderator are taken to be temperature dependent.⁶ In addition to the temperature reactivity feedback effects for the fuel, electrodes, moderator, reflector, and the core support plates, the TOPAZ-II reactor model in TITAM incorporates a correlation of the control drums reactivity worth as a function of angular position.⁷⁻⁹

During the system startup, the reactor thermal power is calculated by the reactor kinetics model on the basis of the rotation angle and speed of the drums and the temperature reactivity feedback for the different components in the reactor (fuel, ZrH moderator, TFE electrodes, coolant, reflector, and core support plates).⁷ For a given reactor thermal power, the coolant temperature and mass flow rate are determined from the solution of the system's overall energy and momentum balance equations. Then the EM pump model is used to calculate the pressure rise in the pump. The pump current and voltage are determined from the thermionic emission model for the pump TFEs.⁸ With the pressure rise for the EM pump calculated, the coolant loop thermal-hydraulic model is solved for the new coolant temperature and flow rate. These iterative solution procedures are repeated until conversion is achieved when both the overall energy balance and momentum balance equations of the system are satisfied. After each iteration, the thermal, physical, and electrical properties of the liquid-metal coolant and structure materials are updated.

MODEL VERIFICATION

The TITAM predictions are benchmarked with the use of results of other calculations that had been performed

by Russian scientists¹² and actual experimental data from the TEST facility in Albuquerque, N. Mex.^{1,13,14} The TITAM results of the startup simulation of the TOPAZ-II system were in agreement with reported values of the total temperature reactivity feedback at steady-state nominal power of 110 to 115 kW thermal (1 dollar and 43 cents) and of the rotation angle of the control drums at the end of the reactor startup process (88 to 90°).^{10,15,16}

The predictions of TITAM are also compared with experimental measurements in Figs. 4 to 7.^{9,13,14} These measurements were taken at the TSET facility for the TOPAZ-II, V-71 system that was tested in November 1992 and May 1993 in which electrically heated TFEs were used. In these tests the middle 0.3 m of the active length of the emitter (0.375 m) in the TFEs was uniformly heated with tungsten electrical heaters. The recorded measurements are for steady-state operation at different electrical power inputs to the heaters of the 37 TFEs in the TOPAZ-II reactor core. As Figs. 4 to 7 show, the calculated coolant temperatures were within 15 K and the calculated coolant pressure was within 12% of the measurements. The model predictions of load electric current and voltage were also in good agreement with measurements (Figs. 6 and 7). This agreement between the TITAM version for TOPAZ-II and experimental data of the system verifies the soundness of the modeling approach.

STARTUP PROCEDURE OF TOPAZ-II IN ORBIT

At cold startup, when the B₄C segments in the safety and control drums are facing inward, the TOPAZ-II reactor is 6 dollars subcritical ($k_{\text{eff}} = 0.952$). The startup procedures assumed herein, which may not represent an accurate account of the actual procedures of the TOPAZ-II system,^{5,7} call for the reactor startup to begin by rotating the three safety drums 180° outward, which increases the core reactivity to a negative 4 dollars ($k_{\text{eff}} = 0.968$). Subsequently the nine control drums are rotated 154° outward, at their maximum speed of 1.4°/s, and then inward to 145°. The reactor becomes critical ($k_{\text{eff}} = 1.0$) when the control drums are rotated 125° outward (Figs. 8 and 9).

The control drums are then held in place until the reactor thermal power reaches 5 kW. When this power level is reached, the drums resume their rotation; however, their rotational speed and direction are adjusted to increase the reactor power to a constant rate of 600 W/s until it reaches 35 kW and then at 80 W/s until it reaches 115 kW. At this point the control drums are rotated inward to maintain criticality of the TOPAZ-II reactor.

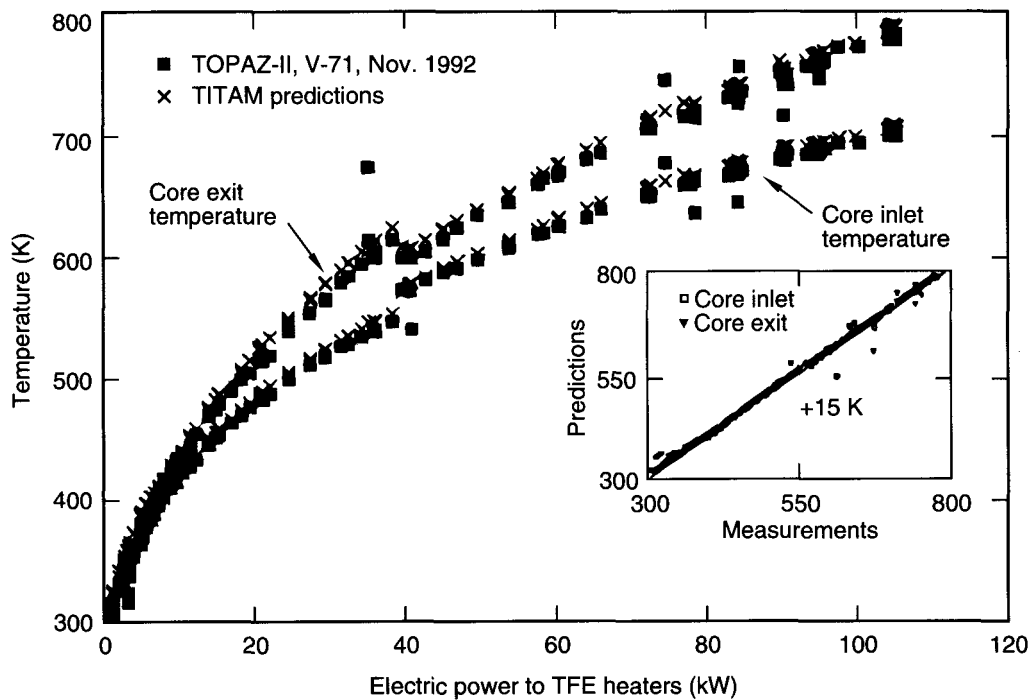


Fig. 4 Comparison of TITAM predictions with measured coolant temperatures in TOPAZ-II, V-71 unit tests in the Thermionic System Evaluation Test Facility. TITAM is Thermionic Transient Analysis Model.

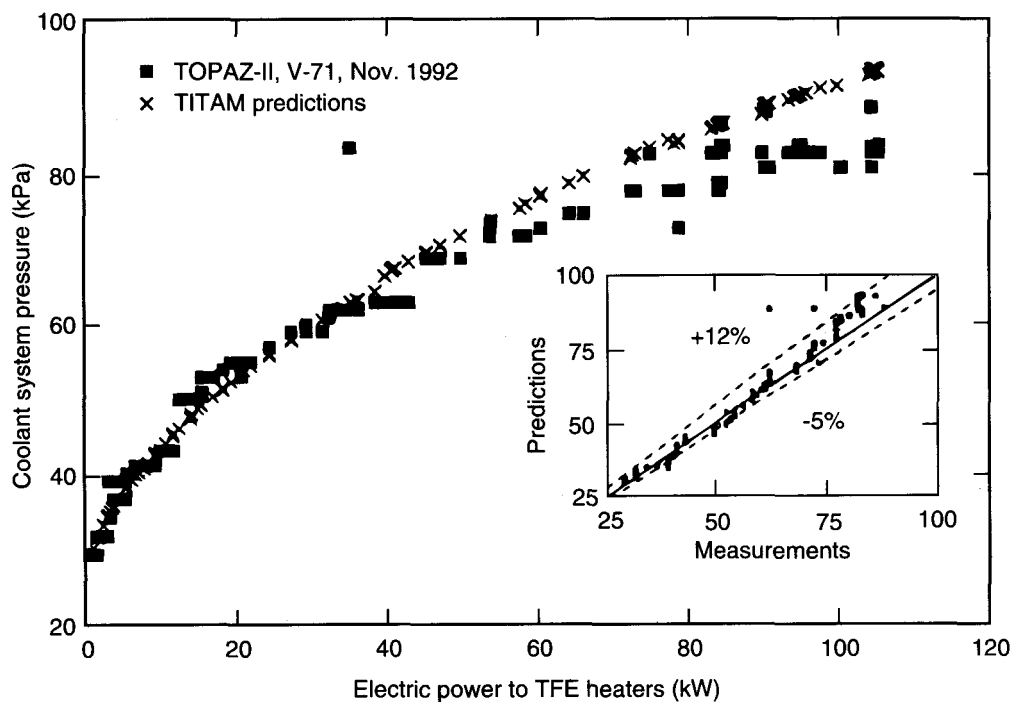


Fig. 5 Comparison of TITAM predictions with measured coolant pressures in TOPAZ-II, V-71 unit tests in the Thermionic System Evaluation Test Facility. TFE is thermionic fuel element; TITAM is Thermionic Transient Analysis Model.

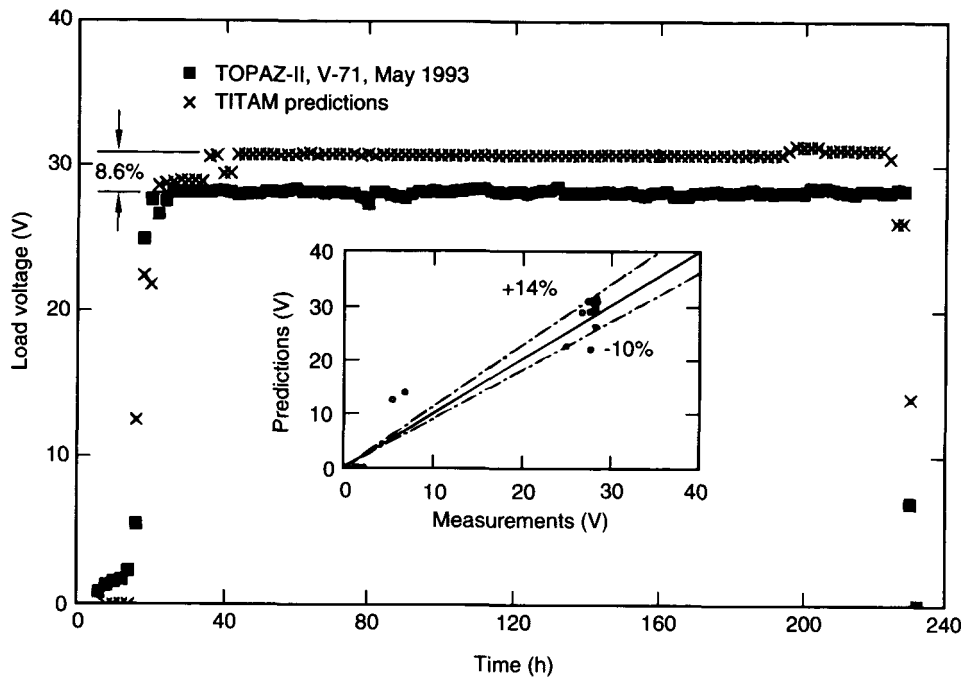


Fig. 6 Comparison of TITAM predictions with measured load electric voltage in the TOPAZ-II, V-71 unit tests. TITAM is Thermionic Transient Analysis Model.

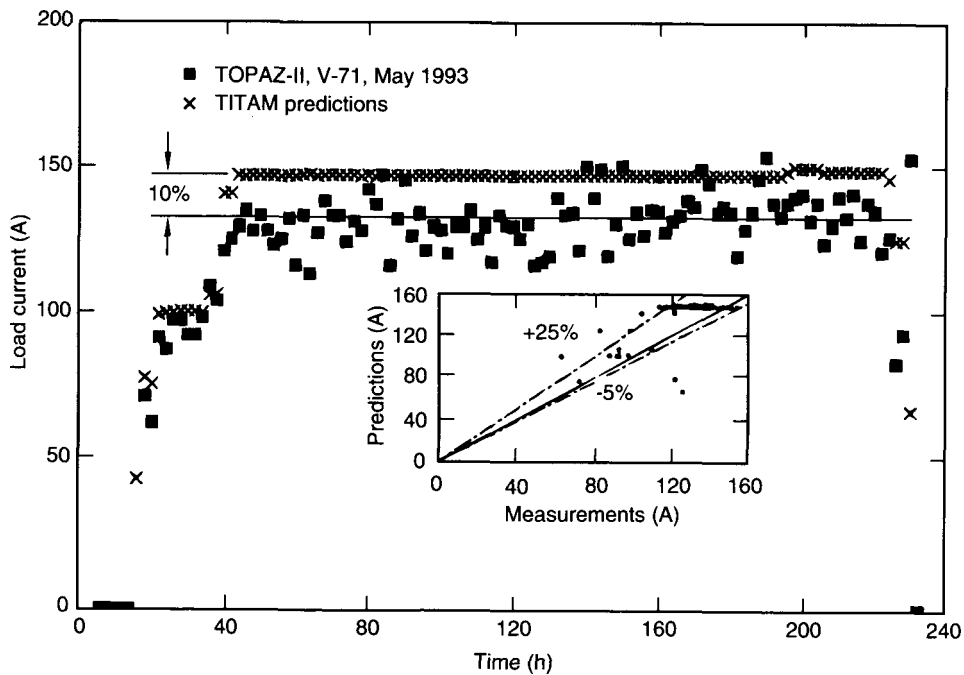


Fig. 7 Comparison of TITAM predictions with measured load electric current in the TOPAZ-II, V-71 unit tests. TITAM is Thermionic Transient Analysis Model.

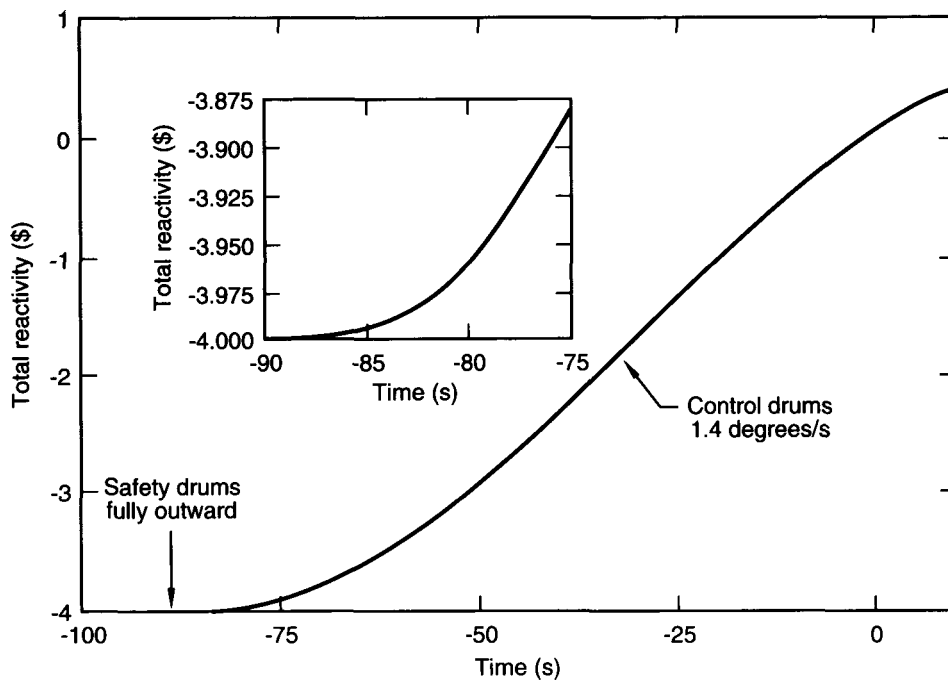


Fig. 8 Total reactivity insertion in the TOPAZ-II reactor during startup simulation.

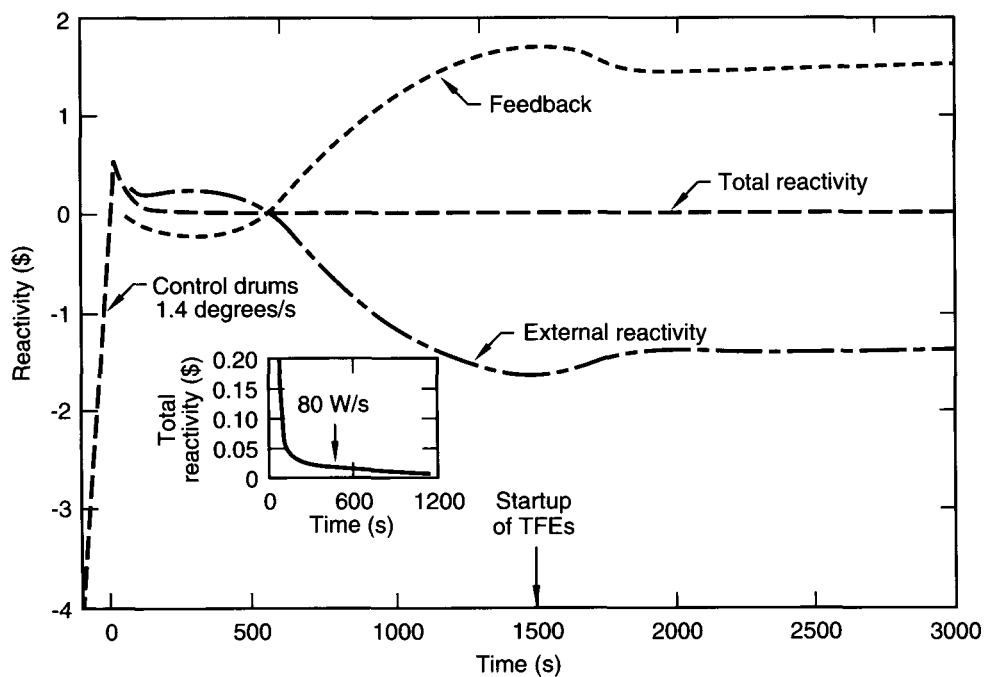


Fig. 9 Calculated changes in reactivity during startup simulation of the TOPAZ-II Space Nuclear Reactor Power System in orbit. TFE is thermionic fuel element.

Figures 9 and 10 show the calculated changes in reactivity and the reactor fission power during startup simulation, respectively, of the TOPAZ-II space nuclear reactor power system in orbit. As Fig. 9 indicates, when the steady-state condition is reached, the total temperature reactivity feedback in the reactor core is about 1 dollar and 43 cents; the angular position of the control drums is about 88° outward.⁷ At this angular position, the total excess reactivity remaining in the reactor core is about 2 dollars and 20 cents, which is used to compensate for the fuel burnup through the lifetime of reactor operation.

RESULTS AND DISCUSSION

The TOPAZ-II reactor radial reflector consists of 12 reflector panels, each housing either a safety or a control drum (Fig. 2). This section investigates the effect of the disassembly of the reflector panels of the TOPAZ-II reactor following a postulated RIA. This accident is assumed to occur because of a malfunction of the drive mechanism that causes the control drums to rotate outward the full 180° range at maximum speed of $1.4^\circ/\text{s}$ and remain out.

During a nominal startup of the TOPAZ-II system in orbit, the reactor becomes critical when the control drums are approximately 125° outward. In Figs. 8 to 15, the zero time corresponds to reactor criticality or to when the rotation angle of the control drums equals 125° . As shown in Fig. 8, the control drums rotate outward for about 90 seconds before the reactor becomes critical.

As shown in Fig. 11, the total external reactivity insertion, 40 seconds after the reactor becomes critical, is approximately 80 cents. However, the corresponding total reactivity in the core is lower (about 75 cents) mostly because of the temperature negative reactivity feedback of the fuel and to a lesser extent because of the electrodes and the core plates (Fig. 13). As demonstrated in Figs. 12 and 13, the disassembly of only 3 of the 12 reflector panels following an RIA would successfully shut down the reactor with little overheating of the fuel.

The reactor fission power peaks at approximately 1.05 MW and then drops rapidly following the disassembly of the reflector panels (Fig. 13). The fuel and the emitter temperatures peak at only about 1410 K, drop rapidly to about 530 K, and decrease slowly thereafter (Figs. 14 and 15). The disassembly of three reflector

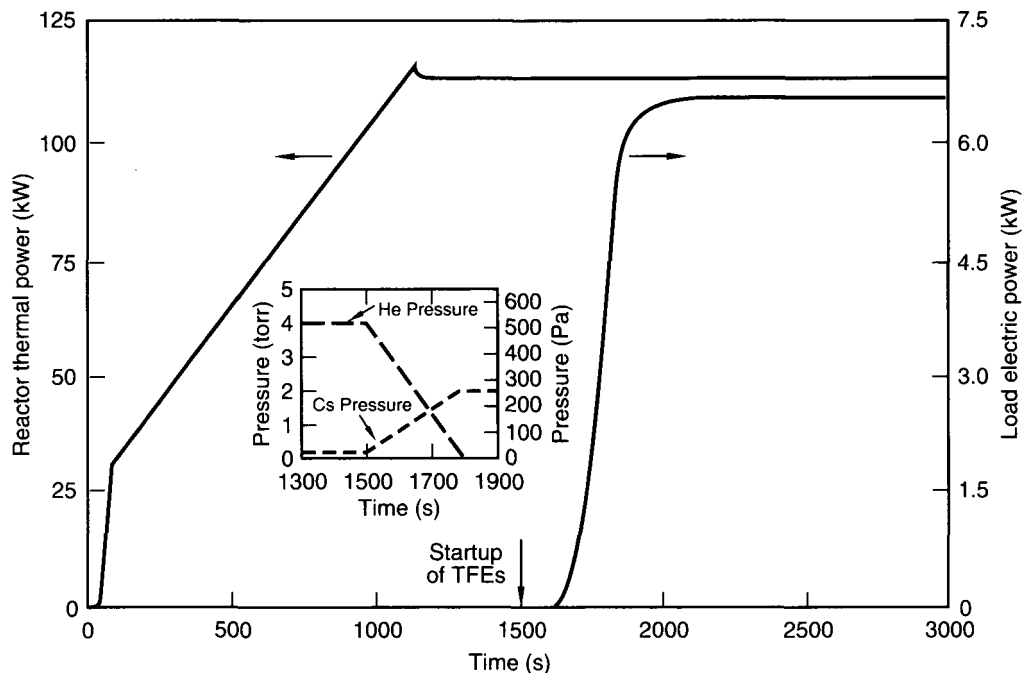


Fig. 10 Calculated reactor fission and electric power during startup simulation of TOPAZ-II Space Nuclear Reactor Power System in orbit. TFE is thermionic fuel element.

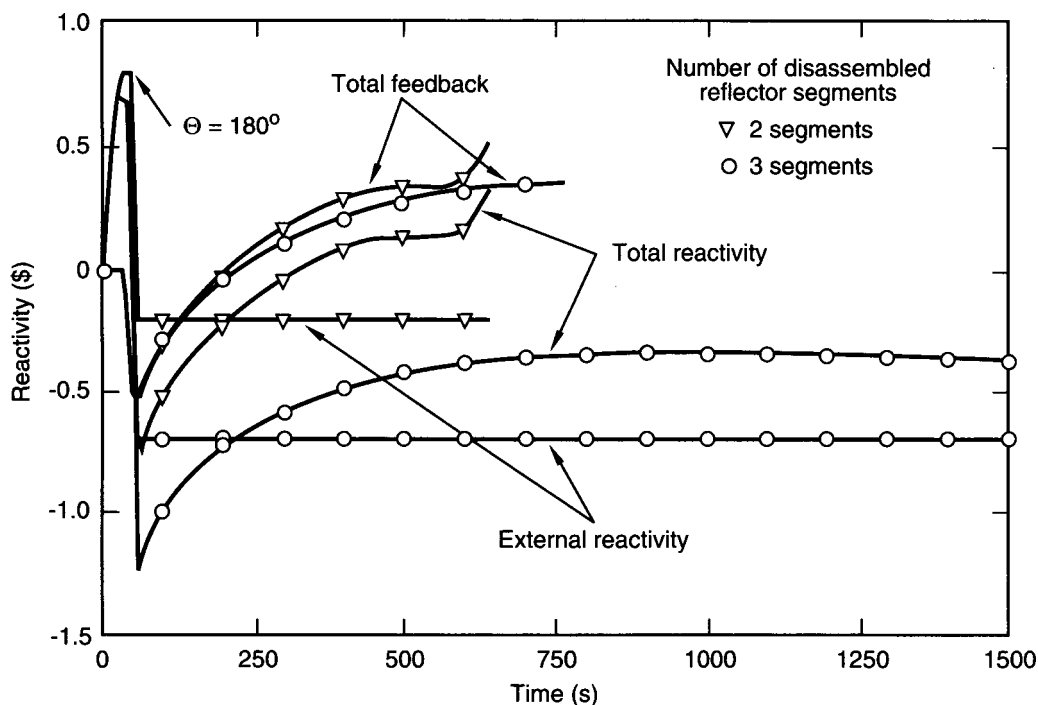


Fig. 11 Effect of the number of disassembled reflector panels on reactor conditions.

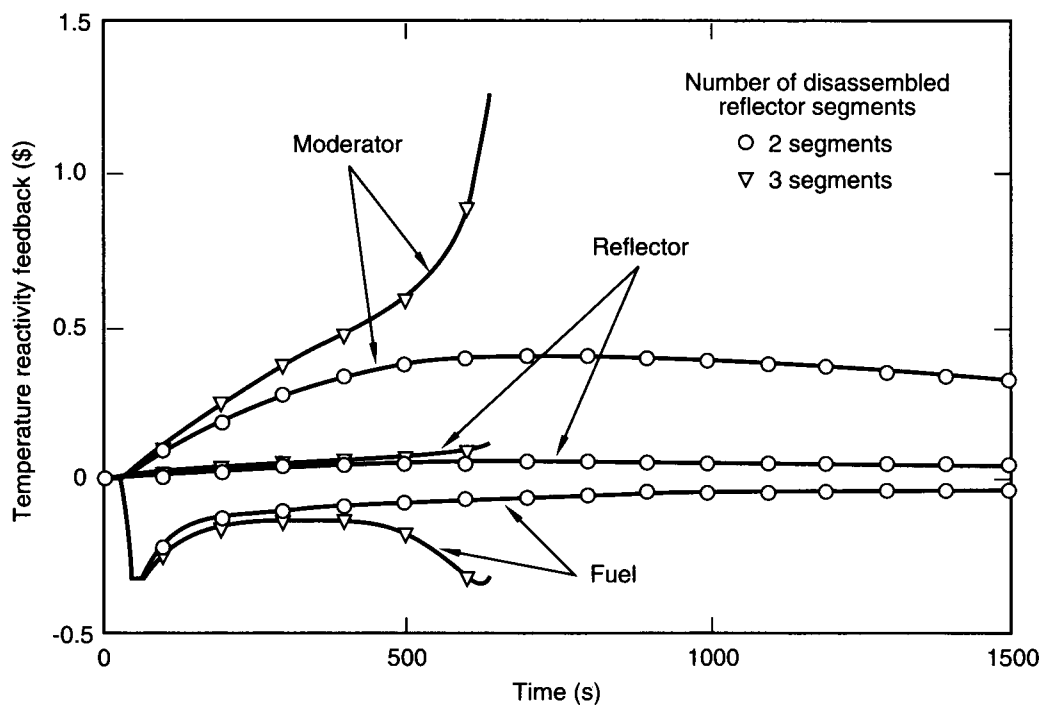


Fig. 12 Effect of disassembled reflector panels on temperature reactivity feedback.

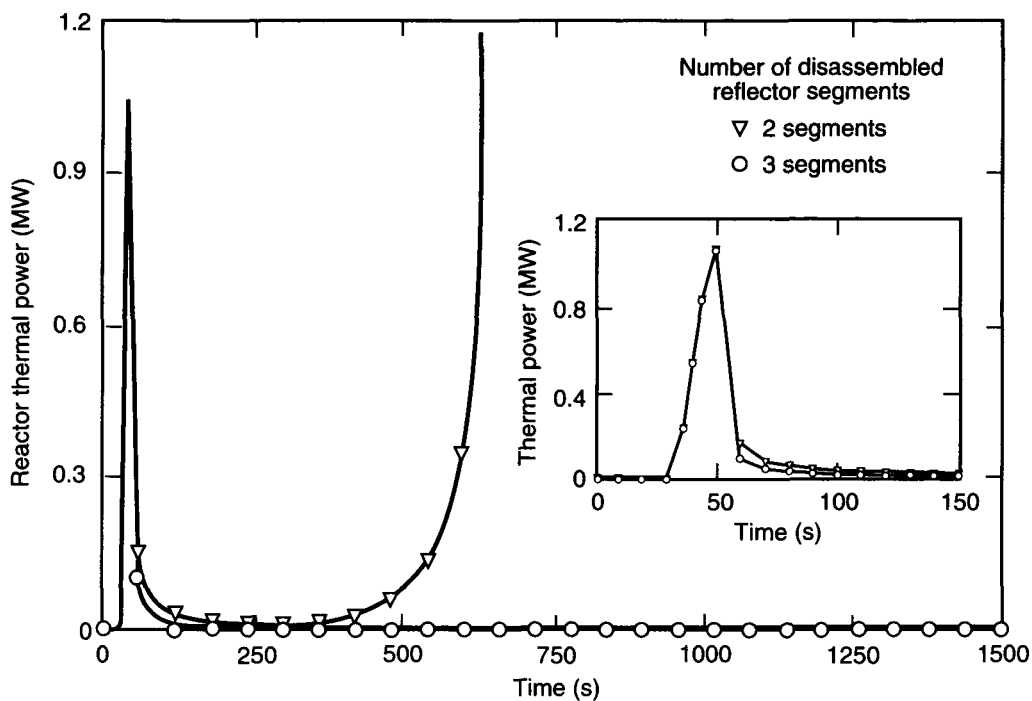


Fig. 13 Effect of disassembled reflector panels on fission power.

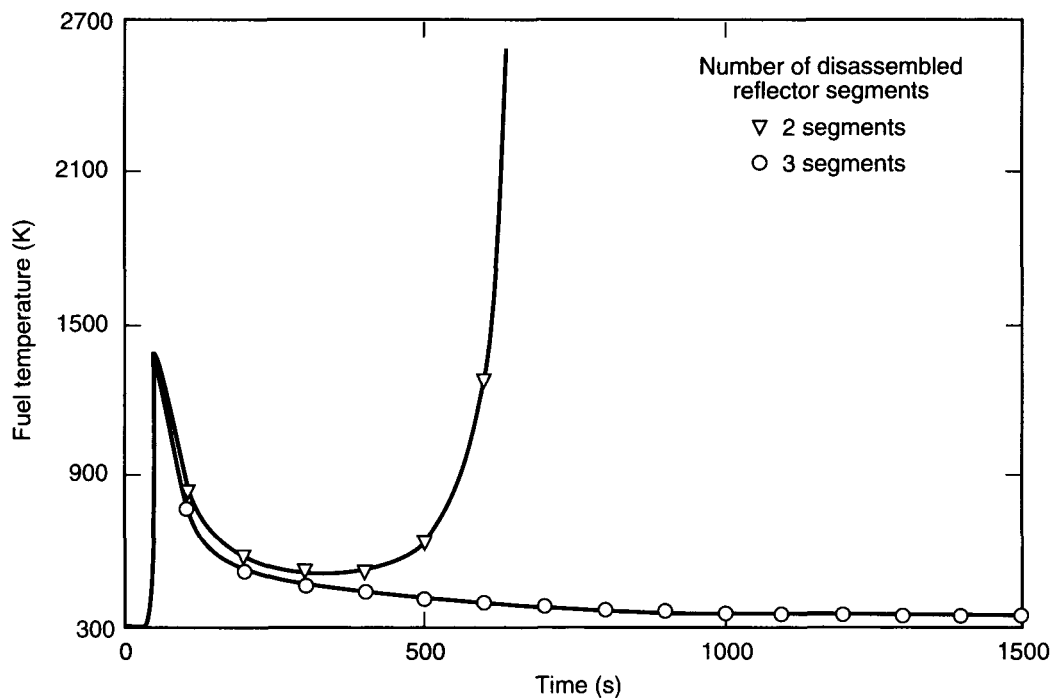


Fig. 14 Effect of disassembled reflector panels on fuel temperature.

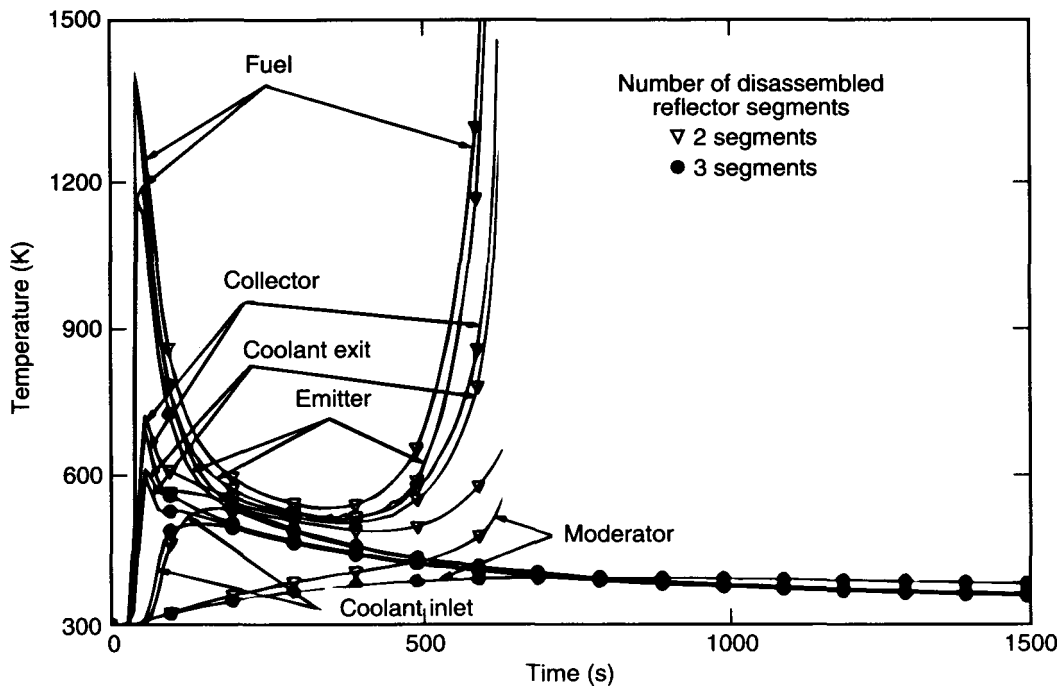


Fig. 15 Effect of disassembled reflector panels on temperatures of reactor core components.

panels inserts a total of negative 1 dollar and 50 cents of external reactivity into the reactor core and thus causes the total reactivity to drop precipitously and reach a minimum of about negative 1 dollar and 25 cents (Fig. 11). Subsequently, the total reactivity increases are mostly caused by the temperature-positive reactivity feedback of the moderator. In approximately 15 min after reactor startup, the total reactivity decreases because of moderator cooling down (Figs. 11, 12, and 15).

The results in Figs. 11 to 15 clearly show that the disassembly of two reflector panels, instead of three, would not prevent a reactivity excursion in the TOPAZ-II reactor and overheating of the fuel and the TFE electrodes. Figure 11 indicates that, following the disassembly of two reflector panels (total external reactivity insertion of negative 1 dollar), the total reactivity in the reactor drops to negative 20 cents and then increases, which causes a reactivity excursion approximately 325 seconds after the reactor reaches criticality. As a result, the fission power (Fig. 13) and the fuel and emitter temperatures (Figs. 14 and 15) increase very rapidly.

The fuel temperature reaches about 2600 K within 11 minutes after the reactor becomes critical. The corresponding collector temperature is about 1450 K, and the coolant temperature at the exit of the reactor core is about 650 K, whereas that of the moderator is slightly less than

550 K. Note that during this time the reactor remains subprompt critical; for example, approximately 10 minutes after the reactor becomes critical during startup, the total excess reactivity only increases to about 50 cents. In these calculations and in those presented throughout the article, the initial temperature of the reactor core is taken to be uniform at 300 K.

CONCLUSIONS

An analysis was performed to determine the effect of the disassembly of the radial reflector panels of the TOPAZ-II reactor following a hypothetical severe RIA. The RIA considered in this article was assumed to occur because of a malfunction of the drive mechanism of the control drums that causes the drums to rotate the full 180° outward at their maximum speed of 1.4°/s and remain out.

Results indicate that the disassembly of only 2 of the 12 reflector panels could eventually cause a reactivity excursion and rapid overheating of the reactor core following a relatively long delay time (more than 10 minutes). Until such time the reactor remains subprompt critical with the total excess reactivity in the reactor core being approximately 50 cents. However, disassembly of only three of the radial reflector panels would successfully

shut down the reactor with little overheating of the fuel and the moderator. These results demonstrate the effectiveness of and the built-in redundancy in the radial reflector disassembly for safely shutting down the reactor in a severe RIA event.

ACKNOWLEDGMENTS

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Standards for High-Integrity Software^a

By D. R. Wallace,^b D. R. Kuhn,^b L. M. Ippolito,^b and L. Beltracchi^c

Abstract: *This article describes a study that examines standards, draft standards, and guidelines (all of which will hereafter be referred to as documents) that provide requirements for the assurance of software in safety systems in nuclear power plants. The study focuses on identifying, for developers of standards, the elements to be addressed in a standard for providing reasonable assurance of software in safety systems in nuclear power plants. The documents vary widely in their requirements and the precision with which the requirements are expressed. Recommendations are outlined for guidance for the assurance of high-integrity software.*

High-integrity software is software that must be trusted to work dependably in some critical function, and if it fails to do so, catastrophic results, such as serious injury, loss of life, or loss of property may occur.¹ Examples include civil aviation, medical devices, nuclear power, weapons systems, and electronic funds transfer. Although we rely on computerized systems in every aspect of living, we are not always sure of the software. Examples of catastrophes, past and potential, can be found in "Risk of the Year" presentations at Computer Assurance (COMPASS) conferences and in major research reports.²⁻⁴

In the nuclear industry, as in many industries, developers and customers need a standard framework of

requirements for software development and assurance that the software of critical systems is of high integrity. Many organizations (e.g., industry associations and international standards organizations) are developing standards to serve this purpose. In this study we list some of the available documents and identify their strengths and weaknesses. We have developed a set of criteria to enable us to identify key characteristics of each document and to determine how well a given document satisfies the criteria for each characteristic. Although this study is restricted to software issues, we also examine the relationship between the standards for software life-cycle activities and those for system life-cycle activities. Additional research may be needed to address requirements for the relationships between software and other system components.

Although the documents we chose to study (see Table 1) vary widely (in scope, life-cycle coverage, and quality coverage), we were able to extract some common approaches in engineering practices and assurance requirements. From the findings, we propose a set of topics, with basic requirements, as a base document on which to develop a standard for the assurance of high-integrity software for use in safety systems in nuclear power plants. The complete study is reported in NUREG/CR-5930, *High Integrity Standards and Guidelines*, and NIST 500-204, *High Integrity Software Standards and Guidelines*.^{5,6}

In the sections that follow, we provide a description of the study, including the questions to be answered and the criteria used to examine the documents shown in Table 1;

^aThe opinions expressed in this article are those of the authors and do not necessarily reflect the criteria, requirements, and guidelines of the U.S. Nuclear Regulatory Commission.

^bNational Institute of Standards and Technology, Gaithersburg, Maryland 20899.

^cU.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

an overview of the analysis of the documents; a summary of the findings; and some recommendations for developers of standards for high-integrity software. The complete study is provided in NIST 500-204.

STUDY OF STANDARDS

We have examined many current national, international, or industry-specific standards, draft standards, draft revisions, and guidelines that address requirements for software assurance. We selected those listed in Table 1 for a detailed study and refer to them generally as documents and specifically by the acronyms listed in the table. Some documents have been developed for the nuclear industry; the remaining documents are intended for large critical systems.

We are interested in two questions: (1) Is there a single document that will provide reasonable assurance of high-integrity software? and (2) Can the customer determine that the developer has met the requirements of the document? We did not expect a single document to fully address every topic; rather, our purpose was to determine how well a document satisfies requirements for any topic it claimed to cover. Our intention is to understand how well the best available guidance from each document might collectively support the assurance of software in nuclear power-plant safety systems.

We developed a set of topics (shown in Table 2 and discussed below) on which to base our analysis. For these topics, we identified detailed criteria necessary for a reasonable standard. The topic list and criteria are not necessarily complete but are based on the research of

Table 1 Documents Used in the Study

Acronym	Number and title
ANS7432	ANSI/IEEE-ANS-7-4.3.2-1982, <i>Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations</i> , American Nuclear Society, 1982.
CATEGORY	<i>Guideline for the Categorization of Software in Ontario Hydro's Nuclear Facilities with Respect to Nuclear Safety</i> , Revision 0, Nuclear Safety Department, June 1991.
DLP880	DLP880, (Draft) Proposed Standard for Software for Computers in the Safety Systems of Nuclear Power Stations (Based on IEC Standard 880), D. L. Parnas, Queen's University, Kingston, Ontario, March 1991.
EWICS2-1	F. J. Redmill (Ed.), <i>Dependability of Critical Computer Systems 2</i> , Chapter 1, "Guidelines to Design Computer Systems for Safety," European Workshop on Industrial Computer Systems Technical Committee 7 (EWICS TC7), Elsevier Science Publishers, 1989.
EWICS2-2	Ibid., Chapter 2, "Guidelines for the Assessment of the Safety and Reliability of Critical Computer Systems."
EWICS2-3	Ibid., Chapter 3, "A Questionnaire for System Safety and Reliability Assessment."
EWICS2-4	Ibid., Chapter 4, "A Guideline on Software Quality Assurance and Measures."
EWICS2-5	Ibid., Chapter 5, "Guidelines on the Maintenance and Modification of Safety-Related Computer Systems."
IEC880	IEC 880, <i>Software for Computers in the Safety Systems of Nuclear Power Stations</i> , International Electrotechnical Commission, 1986.
IECSUPP	45A/WG-A3(Secretary)42, (Draft) <i>Software for Computers Important to Safety for Nuclear Power Plants as a Supplement to IEC Publication 880</i> , International Electrotechnical Commission Technical Committee: Nuclear Instrumentation, Subcommittee 45A: Reactor Instrumentation, Working Group A3: Data Transmission and Processing Systems, May 1991.
NPR6300	NPR-STD-6300, <i>Management of Scientific, Engineering, and Plant Software</i> , Office of New Production Reactors, U.S. Department of Energy, March 1991.
P1228	P1228, (Draft) <i>Standard for Software Safety Plans (IEEE Working Group)</i> , Institute of Electrical and Electronics Engineers, July 19, 1991.
RTCA178A	RTCA/DO-178A, <i>Software Considerations in Airborne Systems and Equipment Certification</i> , Radio Technical Commission for Aeronautics, March 1985.
SOFTENG	<i>Standard for Software Engineering of Safety Critical Software</i> , Rev. 0, Ontario Hydro, December 1990.

Table 2 Criteria Template

Levels of criticality/assurance	
Life-cycle phases	
Documentation	
Required functionality	
Engineering practices	
Assurance activities	
Software verification and validation (V&V)	
Software quality assurance (SQA)	
Software configuration management (SCM)	
Hazard analysis	
Project planning and management	
Procurement concerns	
Presentation	

existing standards and guidelines (shown in Table 3) related to high-integrity systems and the experience of the authors.

Levels of Criticality/Assurance

Some standards have established software requirements on the basis of the consequences of system failure. The most serious consequence is usually considered to be loss of life and is assigned the highest level of criticality. Other levels of criticality take into account how serious a failure would be relative to the completion of the task for which the system is responsible and how devastating the failure would be relative to destruction of property and environment, injuries, and other losses. We looked for distinctions in the requirements of a document on the basis of levels of criticality. This includes requirements according to levels of criticality not only for the principal system but also for its support software and for software used to develop and assure its support. The assumptions of the documents in which criticality levels are defined are that the most critical systems should have the most rigorous software standards and practices.

Life-Cycle Phases

Some documents have been developed for the life cycle of the entire system, whereas others begin with the development of software requirements and do not fully address integration of software within the total system. Because this study focuses on software, for comparison purposes we used only the software life cycle and looked for activities related to software: software requirements, software design, software code, software integration and

test, software installation, and software maintenance. We use the phases only to identify the scope of each document. We are not concerned whether a document specifies a particular life-cycle management (e.g., waterfall or spiral). Although we believe that some activities in software development and assurance need to be performed at certain times in the life cycle (e.g., system test planning during requirements), in general we do not make judgments on life-cycle management. Our priorities center on the activities themselves and how well a document addresses a particular phase. By identifying documents that address partial life cycles, we may be able to determine which documents, or parts of them, may be used together.

In documents dealing with the system life cycle, it is sometimes difficult to know when a requirement is imposed on the software or takes effect only after integration of the software with system components (e.g., configuration management). Again, this study focuses on software-related activities. We also checked that the documents made a clear distinction between requirements for software components and system components.

Documentation

Although we concentrated on *software* documentation, in some cases it was necessary to consider a document's requirements for the system requirements specifications because those specifications levy requirements for software. We considered the following types of questions:

- How thorough are the document's requirements for specific documentation?
- Does the document specify the content that must be described in the documentation? Or does it specify the description of elements of the content?
- Does the document provide a quantified description of attributes that should be present in the documentation (e.g., rules for maintainability, consistency)?
- Is a checklist included?
- Do the requirements for the documentation specify required functionality or engineering practices?

Required Software Functionality Against Hazards

Critical systems must continue to operate despite errors and component failures. To help ensure this, special software functions are often included to detect, tolerate, override, or recover from failures or to prevent execution of unintended functions. Special software

functions should be considered in a standard for safety-critical software. Examples include prompts that query the operator as to whether or not a keyed-in command should actually be executed, and the software used in telephone switching systems where, typically, 50% or more of the software is devoted to error detection and correction. Such functions should be considered in a standard for safety-critical software. Not all the functions listed in the template are essential in all systems, but an evaluator should look for the use of these functions or be aware of reasons they are not needed in a particular system.

Software Engineering Practices

A standard for safety-critical software should give guidance on software engineering practices that contribute to high integrity. Certain engineering practices can either contribute to or detract from the safety and reliability of a system; for example, systems constructed from modules that each perform a single, well-defined function are likely to be more reliable than those where modules perform a mixture of functions (e.g., both control and data input/output). Choice of programming language is another example. Systems written in assembly language

Table 3 Reference Documents in the Study

Acronym	Number and title
ANSI04	ANSI/ANS-10.4-1987, <i>Guidelines for the Verification and Validation of Scientific and Engineering Computer Programs for the Nuclear Industry</i> , American Nuclear Society, May 13, 1987.
ASMENQA2	ASME NQA-2a-1990, <i>Quality Assurance Requirements for Nuclear Facility Applications</i> , American Society of Mechanical Engineers, November 1990.
FIPS101	FIPS 101, <i>Guideline for Life-Cycle Validation, Verification, and Testing of Computer Software</i> , U.S. Department of Commerce, National Bureau of Standards, June 6, 1983.
FIPS132	FIPS 132, <i>Guideline for Software Verification and Validation Plans</i> , U.S. Department of Commerce, National Bureau of Standards, November 19, 1987.
FIPS1401	FIPS 140-1, <i>Security Requirements for Cryptographic Modules</i> , U.S. Department of Commerce, National Institute of Standards and Technology, May 2, 1990.
IDS0055	Interim Defence Standard 00-55, <i>The Procurement of Safety Critical Software in Defence Equipment</i> , Pts. 1 and 2, Ministry of Defence, April 5, 1991.
IEEE828	ANSI/IEEE Std 828-1983, <i>IEEE Standard for Software Configuration Management Plans</i> , Institute of Electrical and Electronics Engineers, 1983.
IEEE830	ANSI/IEEE Std 830-1984, <i>IEEE Standard for Software Requirements Specifications</i> , Institute of Electrical and Electronics Engineers, 1984.
IEEE1012	ANSI/IEEE Std 1012-1986, <i>IEEE Standard for Software Verification and Validation Plans</i> , Institute of Electrical and Electronics Engineers, November 14, 1986.
IEEE1058	ANSI/IEEE Std 1058.1-1987, <i>IEEE Standard for Software Project Management Plans</i> , Institute of Electrical and Electronics Engineers, 1988.
IEEE7301	ANSI/IEEE Std 730.1-1989, <i>IEEE Standard for Software Quality Assurance Plans</i> , Institute of Electrical and Electronics Engineers, October 10, 1989.
ISO9000	ISO 9000, <i>International Standards for Quality Management</i> , May 1990.
ITSEC	ITSEC, <i>Information Technology Security Evaluation Criteria (ITSEC), Provisional Harmonised Criteria</i> , ECSC-EEC-EAEC, Brussels, Luxembourg, 1991.
NIST180	NIST Special Publication 500-180, <i>Guide to Software Acceptance</i> , U.S. Department of Commerce, National Institute of Standards and Technology, April 1990.
NIST190	NIST Special Publication 500-190, <i>Proceedings of the Workshop on High-Integrity Software</i> , Gaithersburg, MD, Jan. 22-23, 1991, U.S. Department of Commerce, National Institute of Standards and Technology, August 1991.
SAFEIT	<i>SafeIT</i> , Vols. 1-2, Interdepartmental Committee on Software Engineering, ICSE Secretariat, Department of Trade and Industry, London, June 1990.

tend to be much more error prone than those developed in modern high-level languages, such as Pascal, C, and Ada.⁴

Rigid rules for the use of specific software engineering practices are not always appropriate. In some cases it may not be possible to develop software for a particular application in a way that is normally considered good engineering practice in other applications; for example, it is considered good practice to separate critical functions from the rest of the system, placing critical functions in a small module that can be more readily analyzed than a large system. It has been argued, however, that some safety-critical systems, those in which safety concerns are present in all functions, cannot be built in this way. It is essential that developers use established good practices as much as possible and explain cases where established practices were not used.

Software engineering practices should be considered in all projects, even if not all techniques are appropriate for all projects. The following major types of software engineering practices are considered in this review:

- Formal specifications
- Component isolation
- Modularity
- High-order languages
- Deprecated programming practices (e.g., floating-point arithmetic)
- Quality attributes (with quantified definition)

Assurance Activities

Assurance activities locate problems in the development processes and products and provide evidence on how well the software complies with its specifications. The activities include software verification and validation (V&V), software quality assurance (SQA), software configuration management (SCM), and hazard analysis. Many of the documents we examined address the system life cycle; in our study we needed to be careful whether system or software activities were addressed; for example, system configuration management may well mean that software configuration management during software development is not required.

Two standards, taken together, ANSI104 and FIPS132, which reference IEEE1012, have comprehensive requirements for software V&V. The criteria for software V&V are based on these standards. The criteria for SQA are derived from IEEE7301. This standard relies heavily on the existence of project documentation. Although the thrust of SQA is *product* assurance, process assurance and process improvement are also important. Although

these topics are mentioned under "Procurement Concerns," they are outside the scope of this study. The criteria for SCM are taken from IEEE828, a standard that is presently serving as a foundation document for an ISO/IEC JTC1 SC7 (Software Engineering) working group in developing a standard on SCM.

A prerequisite for any critical system is an analysis of the hazards or threats that the system must protect against; for example, a power plant safety shutdown system must continue to function even during a power failure. Although we were mostly concerned with hazard analysis applied to software, software hazard analysis (e.g., software fault-tree analysis) is an integral part of system hazard analysis—i.e., software hazard analysis considers the relationship of software hazards identified from system hazard analysis and also potential hazards from software algorithms. Both system and software hazard analyses should be conducted to ensure that all hazards have been identified.

Project Planning and Management

Well-defined project management procedures are as important for the development of high-integrity software as they are for any quality product. The documents reviewed are broad in scope and should contain some requirements on how the development of software will be planned, managed, and monitored. Although the criteria on the template are sparse, we also considered requirements from IEEE1058 and guidelines developed for another federal agency.

Procurement Concerns

Some documents address concerns of the customer; for example, the customer of a system may have concerns about the people who are building and evaluating the system. Are they capable? Should evaluators be independent of the vendor? What should their training plans look like? Do the companies have a quality management policy? Some documents address the assessment of both qualifications of the vendor and of the vendor plans for remaining qualified. Another procurement issue involves the use of automated support to build and verify the system.

Presentation

One of the major problems with using a standard and verifying compliance with it is that all too often the "requirements" of the standard are specified in a disorderly, ambiguous manner. Different categories of requirements are often specified in documentation requirements;

for example, we prefer to see required functionality and required engineering practices stated separately rather than in the documentation requirements for a specific process (e.g., design).

FINDINGS OF THE STUDY

Levels of Criticality/Assurance

Most of the documents reviewed for this report have addressed levels of criticality. Canadian documents DLP880 and SOFTENG do not cite levels of criticality because DLP880 implicitly assumes it is addressing critical software, and SOFTENG is for safety-critical systems, the highest level of requirements. The purpose of the third Canadian document, CATEGORY, is to provide guidance on classifying software according to the consequences of failure, but it does not associate software engineering practices with those categories.

IEC880 makes no distinctions concerning assurance needs in the main sections of the document, and, although Appendix B of IEC880 identifies recommendations for design and programming practices by three levels of priority or importance, no guidance is given on a definition of priority or importance. One purpose of IECSUPP is to clarify and supplement IEC880; perhaps the final IECSUPP will provide guidance on levels of criticality. The draft does this only in specifying diversity requirements, depending on reliability requirements. Although ANS7432 neglects to address levels of assurance, it does require that the tools used for verification have quality assurance activities on them commensurate with their importance to the verification process. IDS0055 is written for the highest level of criticality and very strongly states that all support tools for safety-critical software must be at the same assurance level as warranted by results of hazard analyses on them relative to the safety-critical software.

EWICS2-1 states that the design constraints should be associated with the level of criticality. In EWICS2-2, seven levels of criticality are related to types of systems and values for attributes like unavailability and failure probability. Although the questionnaire in EWICS2-3 provides a table of complexity factors, including the scope of safety considerations, the outcome is not associated with specific practices or assurance techniques. NPR6300 classifies software according to the hardware, function, or activity it supports.

RTCA178A discusses three levels of criticality, and these levels must be addressed in the certification plan for the software. This document is currently undergoing

revision; draft version 4 defines five levels of criticality and provides guidance for determining the level of a system.

There is some disagreement within the software engineering community with respect to levels of assurance and the requirements that are appropriate for each level. Answers to the following questions should help the community decide on the number of levels:

- How many levels of assurance are necessary, and what criteria determine what levels are necessary?
- What activities should support each level?
- Will these activities be industry-specific and/or quality attribute-specific?
- Can significant differences between assurance at each level be demonstrated? Can costs at the next highest level be justified relative to the degree of assurance of each level?
- Can it be shown what costs vendors incur in organizing and building to several levels of assurance rather than selecting only one level?

Life-Cycle Phases

We use life-cycle phases to provide a frame of reference to determine whether a document has requirements for necessary processes, including those for assurance. Some documents are special-purpose documents and address those parts of the life-cycle phases which they affect; for example, the categorization document, CATEGORY, is concerned with procedures and guidelines for categorizing software; the categorization is assigned at the initiation of a project. EWICS2-1 provides guidelines for the design of safety-critical systems and does not provide guidance on other life-cycle issues. The primary concern in EWICS2-1 is the process of design at the system level with some guidance on software considerations.

Some of the documents—IECSUPP, SOFTENG, and RTCA178A—deal strictly with requirements for the software life cycle. They do place the software phases in context with system phases, however. DLP880 and ANS7432 also address system requirements and integration of hardware and software. The EWICS2 documents are concerned with the system level but contain specific references to software. Although IDS0055 provides rigorous requirements for software engineering practices, in other aspects (e.g., SCM) it seems to shift its emphasis to system, even though its title refers to software. In many of the documents, there is often confusion as to whether system or software is the focus of activity.

For safety systems in nuclear power plants in which software is embedded and must always be related to the

system, the following issues on life cycle are especially important:

- Does the document relate software activities to system requirements?
- By treating software as part of the system, does the document omit necessary emphasis on software (e.g., are configuration management requirements at the system level only)?
- By combining requirements of the documents, are all life-cycle processes covered adequately?

No single document provides sufficient requirements to satisfactorily address the first two questions. The combination of the documents provides coverage, at least minimally, for the life-cycle processes. Although IEC880, in spite of its problems with presentation, comes close, it does not address project management. Other documents do better in other areas (e.g., NPR6300 on reuse and corrective action). Although ANS7432 does address the software-system relationship, overall its requirements for software for all life-cycle processes are either minimal or nonexistent. If taken as a whole, the EWICS2 documents address (system) design and maintenance but provide little guidance on other aspects of the software life-cycle processes. EWICS2-4 and SOFTENG are the only two documents to address quality attributes and measures for them. With respect to maintenance, IEC880 and EWICS2-5 provide more guidance than the other documents.

RTCA178A, which is currently under revision, provides rather generic requirements for most life-cycle processes; the revision will probably be oriented more toward processes, not phases, and may contain rigorous requirements.

Documentation

The requirements for documentation range from a simple statement for each type of document to a complete description of the quality attributes of a document. ANS7432 is terse: completeness, consistency, and documentation standards are implied. The most complex set of documentation requirements is in DLP880 where documentation is to be written in formal specification languages.

Only one document, SOFTENG, provides rigorous guidance on the quality attributes that should be inherent in documentation. For each document, criteria are identified for each required quality attribute.

One of the features of several documents, especially IEC880 and SOFTENG, is that documentation requirements include requirements for the software itself; for

example, IEC880 specifies documentation that describes features of system modules. The requirement that a module should have a single well-defined function is a requirement for the design, not the documentation. The design of modules is an intellectual activity for designers to structure the system with the best possible design for the system's operational capability and assurance according to design requirements. The documentation should reflect, not require, the design resulting from this intellectual activity, which must take into account requirements for the design. In other cases, functional requirements were hidden in documentation requirements. Standards that make the distinction between documentation requirements and those for engineering practices and functionality should benefit developers. The distinction should also facilitate the auditor's task of verifying compliance with a standard. The documentation requirement may be that the engineering practices should be documented separately, as, for example, in IDS0055's requirements for a "Code of Design Practices."

In most of the reviewed documents, separate documentation is specified for each life-cycle phase or process. An exception is RTCA178A, which treats the software development and verification plan as one document.

Required Software Functionality Against Hazards

IEC880 and EWICS2-3 include lists of software functions that can be used to counter specific hazards. Of these two, the checklists in EWICS2-3 are the more comprehensive. IEC880 contains annotations indicating what each function is "good for" and "good against," but these are generally obvious, so the annotations provide little useful guidance. For example, the annotation for Retry Procedures indicates that retries are useful against sporadic hardware faults; range checking of variables is said to help guard against "yet undetected errors."

The inclusion of checklists of software functions to guard against hazards is helpful in a standard, but it is probably not appropriate to mandate specific functions when a standard covers a broad category of systems. Some functionality that is considered essential for all nuclear safety systems (for example, range checking) might be required, but, in general, some functions may not be appropriate for all systems.

Software Engineering Practices

Software engineering practices are techniques that help prevent errors during system construction or help ensure integrity in operation. An example of the first type

is the use of formal specification languages, and an example of the second type is the use of modularity.

The documents reviewed vary enormously in their recommendations regarding software engineering practices. Most contain at least some guidance on good practices with the two exceptions of RTCA178A and NPR6300. Although EWICS2-4 and EWICS2-5 do not address engineering practices, the other EWICS2 documents provide comprehensive coverage of recommended practices. The following summaries are given approximately in order of consensus among the documents.

- *Modularity and critical component isolation*—The software engineering practices cited by most of the documents are the use of modularity in design and the isolation of critical components.

- *Programming language*—Several documents state principles for programming languages. The consensus is for use of high-level languages (e.g., C, FORTRAN, and Ada) rather than assembler and for languages that support automatic checking of data types and function arguments. For example, Ada or C++ will warn if a function is called with an integer argument when it is expecting a character string. Other considerations may include the value of using languages that support strong data typing and the use of structured programming (the use of restricted control structures rather than arbitrary branching).

- *Formal methods*—Over the past decade, there has been increasing interest in the use of formal methods (i.e., the use of mathematical logic and related areas of mathematics to specify and model the behavior of software). Formal methods are required by only two of the documents reviewed (DLP880 and IDS0055). Another, the supplement to IEC880 (IECSUPP), says that "formal methods should be considered for the highest requirement of safety importance." EWICS2-3 gives preference to the use of formal specifications over informal ones but does not require the use of formal methods. IEC880 notes only that "a formal specification language may be a help to show coherence and completeness of the software functional requirements." A trend toward greater reliance on formal methods is evident in the documents we reviewed. DLP880, IDS0055, and IECSUPP were 1991 drafts; EWICS2-3 was written in 1989, and IEC880 in 1986.

- *Documentation of software engineering practices*—The documents reviewed give adequate treatment to most aspects of software engineering practices except in the area of formal methods.

- *Quality attributes*—Only SOFTENG and EWICS2-4 provided either specific requirements or measures for

quality attributes like completeness, consistency, and maintainability.

Assurance Activities

In the nuclear power industry, as in many other industries, software is one component of a company's business. At the top management level, the view is of the whole, not a part. *System* configuration management and *system* validation are the engineering concepts that make sense to executives of manufacturing companies. For software companies, executives think in terms of *software* configuration management and *software* validation. The difference is not trivial and has caused much misunderstanding in the development of standards. Software is deeply embedded in systems in which software cannot fully stand alone. In these systems nonsoftware components are often not only plug-in but are also built to precise, accredited standards. Configuration management and testing of these components during their development are expected activities. Software should be treated similarly. This has not been possible in the case of testing. First, software systems are usually unique for each system in which they will be embedded. There may be no precise set of validated specifications. Second, their full functionality can only be simulated and cannot be tested in real time.

In this study we concentrated on how well the documents provide assurance of the software. Few of the documents are focused entirely on software. We found ourselves second guessing whether system-level requirements applied at the software development level or applied only at the point when software was integrated with the system. If we are unsure of requirements, how can auditors check for exact compliance? If more accredited, precise standards for software existed, as they do for other components (e.g., pipes and power cables), then the task of demonstrating compliance would be easy. This study reemphasized for the authors the growing recognition that the software industry must develop more precise standards for software that permit the measurement of its quality.

For the assurance activities, P1228 focuses on safety issues and requires specific assurance activities. Under P1228, all documentation for assurance activities may serve as special sections of plans for those activities (e.g., safety requirements for the software V&V plan, for the SQA plan, and for the SCM plan). The assumption of the P1228 draft is that the other IEEE Software Engineering Standards, or similar standards, will be used. For computer security planning, it may be possible to adapt the documentation requirements of P1228.

Software Verification and Validation (V&V). The difference between system and software viewpoints stands out in the documents; for example, ANS7432 is concerned with computer system validation and not particularly with software issues. Software testing comprises software verification; in the software world, this can be confusing because software verification also includes many types of static analysis. Part of the rationale for not treating verification and validation as separate functions in IEEE1012 is to avoid this confusion. The final step of software V&V is the system validation, as in system standards; software V&V consists of these activities applied as the software evolves to assure the internal properties of the software and the external relationships to the system.

DLP880 refers to software verification but actually deals with software V&V. One caution with DLP880 is its assumption that the vendor may produce the verification plan, which is then implemented by an independent team. This is not the only meaning of independent V&V (IV&V); the fullest possible benefit of independence is the independent planning process in which the IV&V brings a different perspective to the types of analysis and test strategies.

Two documents, EWICS2-1 and P1228, focus on the safety requirements; this is acceptable because these documents are intended to augment other more general documents and the intent is to ensure attention to the safety functions. EWICS2-3 should be used by verifiers as guidance in checking features of the software and by auditors to check how well the developers and verifiers have followed guidelines.

IEC880 specifically addresses software verification and is reasonably thorough. There are weaknesses, however. The major weakness is that of presentation; the reader must search several places before finding all the requirements imposed by the standard. Technical weaknesses include a lack of specific requirements for requirements traceability.

Although some test strategies are recommended in several documents, some major strategies have been omitted. For example, IEC880 has long lists of strategies and conditions but omits stress testing. Techniques for error detection (e.g., inspection and testing) are required in several documents, but the analysis of errors to identify common errors or problems with the development process is not mentioned. Error analysis should be a requirement in all V&V standards or sections of standards addressing V&V (or possibly in SQA). Error analysis is important for uncovering a type of error (e.g., misunderstanding of trigonometry) that could appear elsewhere in

system. When common errors are made because of a misunderstanding or a wrong specification, it is important to check other places in the program on the basis of the same assumptions. Otherwise, a potentially critical error could slip through.

Although RTCA178A provides a well-organized set of requirements for software verification (including validation), the software verification assurance matrix is too high level to be truly useful for auditors.

From our review of these documents, including the base documents IEEE1012 and ANS104, we would make the following recommendations for improving standards for software verification and validation:

- There should be a clearer relationship to the system life cycle.
- Practices should be based on levels of criticality.
- Distinct requirements should be spelled out for different types of tests.
- Detailed checklists are needed.
- Standards should address application of V&V when modern development technologies are used (no document addressed V&V for prototyping or expert systems).
- Error analysis should be a requirement in all V&V standards.
- Standards should define the quality attributes for which verification is required.

Software Quality Assurance (SQA). The documents of this study are concerned strictly with SQA of the product, not the vendor processes. Current and evolving SQA standards are addressing process as well as product. When a nuclear customer reviews a particular product, will the customer be concerned about whether a vendor has changed processes in midstream or for the next product? Probably not. For a current review, the customer is more likely to be interested in whether a given product has the required quality level. But when new SQA standards are written, what happens if they require activities for both process and product? Nuclear customers need to study this question to determine if process quality is outside the scope of their reviewers.

Several documents addressed SQA in a general manner. For example, ANS7432 requires that SQA be addressed in the software development plan. IEC880 simply requires an SQA plan. This is insufficient because it does not make clear what is required of the vendors. For audit purposes, the customer must know the following: (1) the minimum set of SQA activities that are to be performed and (2) to what degree SCM and software V&V are included in SQA.

Some documents permit the use of national standards. In most cases the user of a standard needs to ensure specification of an SQA standard in a contract and auditors need to know every SQA standard quite well.

Design and code inspections can be either SQA or software V&V activities. Requirements for inspections were not consistent in the documents. Both ANS7432 and EWICS2-1 provide detailed procedures for SQA of design, and EWICS2-4 addresses SQA entirely.

There is a growing recognition that SQA procedures are needed for existing software programs and for reuse of software modules. NPR6300 provides detailed guidance, and it appears that IECSUPP will address the topic also. The SQA sections of the documents, like those for software V&V, provide little information concerning anomaly reporting, corrective action and follow-up, and error analysis.

Software Configuration Management (SCM).

Software configuration management is another process that is sometimes addressed only at the system level. Although the size of software used in safety systems for nuclear power plants may be small, the critical role of software in safety mandates that SCM be required for all the life-cycle processes and products of such software. IEC880's requirement for system configuration management is not sufficient. ANS7432 and EWICS2-3 simply ignore the topic. Several documents (DLP880, SOFTENG, RTCA178A, and NPR6300) require SCM activities with varying degrees of rigor. The international community (ISO/IEC JTC1 SC7) is presently using IEEE828 as a base document for producing an international standard on SCM. When the standard is approved, the nuclear community in general may consider simply citing this SCM standard directly in contracts and in other guidance that should have an SCM requirement.

Software Hazard Analysis. Typically, the initial hazard analysis is performed from a total-system, environmental perspective, and the results may affect the system and software requirements and design. The results of that analysis and successive hazard analyses should be an input to the software assurance activities. By examining these results, the software experts identify what potential hazards have an impact on the software or may be mitigated or prevented by software. In addition, hazard analysis specific to software should be conducted. There is some debate over whether software hazard analyses should be considered software development or assurance activities. Perhaps the perspective of software assurance may lend itself somewhat better to conducting

software hazard analyses and using system hazard analyses to check the safety impact of the software.

The purpose of CATEGORY is to identify the criticality category of software for the nuclear industry. This document does not define methods for hazard analysis but does provide criteria for criticality to be applied to the results of hazard analyses.

Many of the documents appear to assume that a hazard analysis/criticality assessment has been performed (e.g., IEC880 and DLP880). SOFTENG and RTCA178A indirectly require hazard analyses. P1228 requires safety-related analyses on the results of the system preliminary hazard analysis; the guidance in this draft standard should be required for any software related to large, complex systems.

Project Planning and Management

Most of the documents in this study either do not address project management activities or do so indirectly through other governing principles. For example, requirements on planning for software quality assurance and software configuration management may be considered requirements for project planning. SOFTENG includes requirements for project management in requirements for the Software Development Plan. EWICS2-4 addresses project management through acceptance criteria. The P1228 draft expects a project management plan and requires that the plan be augmented to address software safety issues.

Procurement Concerns

One concern of customers is whether assurance activities should be performed by independent teams. ANS7432 uses a nonbinding statement in the foreword to recommend independence and requires independence of the verification group at the system level. Others, like DLP880 and IEC880, recommend that verification plans be written so that an independent team may implement the plan. IECSUPP suggests complete separation of development and verification teams. EWICS2-1 and EWICS2-2 ask for an IV&V assessment. P1228 also recommends IV&V. SOFTENG asks for independence between development and verification; management of the developers is different from management of the verifiers (but does not require a separate organization). These recommendations present another problem: nowhere is there a standard definition of "independence" and of the tasks of IV&V. In regard to the first issue (a standard definition of "independence"), can the "independent" team be simply another department within a vendor's

organization? What conditions make the team "independent"? In regard to tasks, a nonexclusive list of possible meanings of IV&V duties includes the following:

- The independent team writes all test plans and executes them.
- The independent team performs static analyses on the software design.
- The independent team performs only test execution.

Unless the document clearly specifies a definition of IV&V, requirements for IV&V are ambiguous.

Most of the documents do not specify an assessment of contractor capability, although the EWICS documents do ask for compliance with ISO9000, which requires assessment of contractor's quality system. P1228 requires the software safety plan (SSP) to specify qualifications for the personnel performing software safety activities.

For an auditor to verify vendor compliance to a standard, it is helpful to have a statement of conformance within the standard. Only two of the reviewed documents have firm conformance clauses. SOFTENG states that conformance means all its requirements must be met. Of the five separate chapters in EWICS2, all except one have strong conformance clauses that list specific requirements; for example, EWICS2-1 requires written procedures that identify the existence of activities corresponding to every step of the guideline. EWICS2-3 consists of a questionnaire; a conformance clause is inappropriate.

Several of the documents suggest that the use of preexisting software in a product falls under the requirements of a document. Some also require the same level of assurance for automated development or assurance tools. For example, P1228 states that preexisting software must be in compliance with P1228 and that the verification of support tools depends on the level of assurance of the system. Although IEC880 has requirements on the use of operating systems, it does not require that automatic development and verification aids be tested.

Presentation

The documents in this study have a variety of problems with their presentation. The major problem lies with usage of words to indicate requirements: "shall," "should," "must," "may." Use of the words "shall" and "should" in the same paragraph can be confusing to vendors, assurers, customers, and auditors because "shall" in a standard means "required" but "should" means "desirable but not required." Requirements and

recommendations need to be clearly distinct from one another. Some requirements in several documents [e.g., "the software must be easy to test" (IEC880)] are meaningless because they are not precise enough to objectively test for conformance. An example of language with meaningful constraints on qualities may be found in SOFTENG (e.g., quantified definition of completeness for a software requirements document).

Another concern is that features required to be present in the software, development practices, and descriptions of the software are often specified in requirements for documentation. A mechanism frequently used is to define documentation as a description of a life-cycle process; for example, in IEC880 and SOFTENG, documentation for a software requirements specification often specifies the principles or functions the system must embody or, in the case of software quality assurance, the activities to be performed. Engineering practices are hidden in documentation requirements. Those practices are discussed under "Documentation" in this report. We recommend that standards separate different categories of requirements.

CONCLUSIONS AND RECOMMENDATIONS

No single standard or guideline in this study completely satisfies the evaluation criteria given in Table 2. In almost every case, each document satisfies at least one category. Developers of standards for high-integrity software should include rigorous requirements for engineering practices, required functionality, software project management, documentation, hazard analysis, software verification and validation, software quality assurance, and software configuration management. The language of the standard must not be ambiguous. It is acceptable to cite specific standards to provide requirements for any of the categories (e.g., IEEE1012 and ANSI104 for software V&V).

We recommend that developers of standards for high-integrity software take the strong parts of these documents to build a rigorous guide for assurance of high-integrity software, or, alternatively, to encourage development of a standard on the basis of these documents; for example, DLP880 and IDS0055 have the most rigorous requirements for engineering practices and provide a sound means for developing safety-critical software. Requirements can be extracted from IEC880 for almost every category and P1228 provides requirements for ensuring that safety considerations are given attention at every step of development and assurance; the same concepts could be used for ensuring attention to computer security issues.

Information from all the documents can be used in developing a rigorous standard, but additional concerns must be addressed. One of these is a clear identification of the scope of the standard relative to system and software life cycles. In addition, the requirements for existing software must be addressed to ensure that the software meets the required quality for the safety system. The requirements for software must ensure that the software is always assured relative to its relationship to the system. A standard should either define a practice or specify a standard for it; citing good "software quality" practice is not sufficient because good software quality practice can be interpreted in many ways. Until fundamentals of software engineering practice are rigorously codified in handbooks as in other engineering fields, we recommend that guidance for assurance of high-integrity software either describe all practices or cite specific acceptable standards for them, including at least the topics in Table 2. The standards must be written so that the requirements may be adapted for new technology.

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Design Features

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Adoption of New Design Features for the Next Generation Nuclear Power Reactors

L. S. Tong^a

Abstract. *This article provides a technical basis for establishing a balanced emphasis regarding plant safety, operability, constructibility, maintainability, and economics in the selection of a new nuclear power plant. Energy Cost Differentials and living-probabilistic risk analysis (PRA) are suggested for selecting new reactor designs in this paper. The Energy Cost Differentials represent the anticipated economic benefits obtained from the analyses of living-PRA in the respective stages of design, construction, and operation on the basis of safety, reliability, operability, and availability characteristics of various new designs. Thus the buyer(s) can be assured that his investment in such a new power plant would be safe and profitable.*

The guideline for selection of a new reactor design is suggested in the following steps:

- 1. Set up design goals by the potential buyer(s).*
- 2. Design strategies formulated by the vendor to meet the design goals.*
- 3. Review the vendor's plant performance analyses.*
- 4. Review the proposed control and protection systems, the operation Test Specifications, and the training procedures.*
- 5. Review the designer's safety analysis to ensure the licensability of the plant.*
- 6. Record the evaluation results of above reviews in terms of the evaluated Energy Cost Differentials of each proposed plant for comparison and selection.*

Various conceptual designs for the new reactors have been published,¹⁻¹¹ and the Advanced Light-Water Reactors Utility Requirements Document (ALWR URD)¹² has

been developed and evaluated by the Nuclear Regulatory Commission (NRC) in a formal safety evaluation report. U.S. reactor vendors are diligently seeking the certification of their advanced designs by the NRC. It is time for potential buyers (domestic or foreign) to evaluate seriously the maturity of these new designs and to confirm the improvements in safety, reliability, maintainability, and economy in each of these new designs.

Nuclear Power,¹¹ authored by the Committee on Future Nuclear Power Development and published by the National Academy of Sciences, provides excellent advice on the services and improvements offered by the nuclear industry; however, it does not provide numerical rankings or a critical comparative analysis of the practical technological options for the future development of nuclear power. A major problem in making such comparisons is that there is an unbalanced emphasis in addressing reactor safety and management policy issues. Thus the design goals and design strategies of various reactor developers are varied and not clearly defined.

The safety and management policy issues^{13,14} that require balanced emphasis are (1) reactor safety versus plant economies, (2) accident prevention versus consequence mitigation, (3) standardization versus customization, and (4) initial cost optimization versus energy cost optimization. If an agreement can be reached on the proper balance of emphasis for each of these issues, then guidelines for comparing and ranking advanced designs for new reactors can be established. To be acceptable, the final set of safety and management issue resolutions must be agreed upon by a large cross section of the industry so that the balanced emphasis in the

^aIndependent Consultant, 9733 Lookout Place, Gaithersburg, Maryland 20879.

design strategies will be used jointly by designers, constructors, owners, and regulators in the further development of new reactors.

The objective of this article is to provide a technical basis for establishing a balanced emphasis regarding safety and management policy issues and to propose a set of guidelines for potential buyers to adopt in selection of new advanced designs for the next generation of nuclear power reactors. The technical basis and logic of the selection process are developed in a step-by-step approach.

First, a set of design goals is suggested for adoption by the potential buyer. The overall design goal can be specified as goals of four separate functions. They are mutually complementary, and the emphases of these separate goals are balanced with each other. The goals for the four functions are

- Safety and reliability
- Operability and availability
- Constructibility and maintainability
- Economics

Second, a set of basic design strategies is suggested for vendors to use in achieving the design goals. In practicing these strategies, a consolidated design process is suggested for use in an integrated plant design. The design criteria of systems and components can be developed by using the principle for standardization, whereas the risk of common-cause failures is carefully limited.

Third, for the evaluation of the maturity of the proposed new design features, a collection of new design features proposed by various vendors is tabulated and categorized according to their system functions. For confirmation of the designed performance of the proposed plants, the vendors are requested to provide (1) quality assurance (QA) practice and testing procedures in the design stage and in future fabrication, (2) maintenance plans (short- and long-range) with spare parts supply information, (3) the operating Technical Specifications, and (4) operator and maintenance personnel training programs. The vendors are also requested to demonstrate the design efficacy of their control and protection systems, as well as instrumentation and communication systems, in reviewing their proposed advanced control rooms.

Fourth, vendors are requested to present the data bases and the resulting risks from the safety analyses of the proposed plants and to provide additional safety information for use in obtaining the operating license of the plant at a specific site.

Fifth, a conversion technique is suggested for converting the beneficial influences of the new design features into energy cost reduction. The net reduction of the

energy cost of a proposed plant is then compared with that of other plant designs for optimization. Finally, guidelines for the selection of new reactor designs are presented for implementation.

DESIGN GOALS

The overall design goal is to produce a safe and reliable power plant that can be operated easily at various load levels. It should be constructed in a reasonably short time and be maintained by reasonably simple procedures. Further, its energy generation cost should be competitive with that of other power sources. When a new nuclear power reactor is designed, built, and put into operation, the three principal participants (owner, vendor, and regulator) share responsibilities for the proper execution of the project in addition to their own basic responsibilities. The owner is responsible for the overall management of the project, including operation safety and economics. The vendor is tasked to reactor design, fabrication, and operation readiness, whereas the regulator monitors the entire process to ensure that public safety is safeguarded in design and operation of the plant. Naturally, each participant tends to be more concerned about the part of the plant closely related to his own responsibilities. However, all parties should share a common understanding that good reactor design should be balanced in meeting all the design goals. The three principal participants must work cooperatively to pursue their common design goals. A good example of cooperative work in progress is the NRC's final design approval (FDA) and certification of new reactor plant designs with reference to ALWR URD, which was suggested by the Electric Power Research Institute (EPRI).^{12,15} Final design approval from NRC is an indication of ensuring public safety and should greatly enhance public acceptance of the design.

The first tier of the utility design requirements for ALWRs is outlined in Appendix A. These requirements contain the owner's goals for operating a plant and the vendor's design strategies for meeting the owner's goals. These design goals are set for the designers as well as for the builders and operators because a good design requires a good builder and a good operator to achieve the designed performance. Both the vendor (designer and builder) and the owner (manager and operator) will have to do their jobs well to meet the design goals.

To choose a new reactor design for construction and operation at a specific site, the buyer usually invites the qualified vendors to offer their proposals in an open bid. For the convenience of a buyer in formulating the bid

specifications, this article suggests a set of design goals for buyers to consider. These goals are delineated in Table 1. Most of the details of the four design goals are taken from the utility requirements of the published ALWR URD.^{12,15} However, these requirements are recategorized to clearly distinguish the owner's goals from the vendor's design strategies. Some of the requirements are modified to balance the emphasis of the safety and management policy issues.

In formulating the safety design goal, a set of comprehensive safety principles is used as the prerequisite of a reactor design, and credible tools for safety analysis are suggested for determining the priority of accident prevention vs. consequence mitigation in a plant design. Thus the emphasis of the reactor safety issues can be balanced in the safety design goal. Because the beneficial influences of the safety requirements, both in design and in operation, can be converted into increase of availability and reduction of energy cost, the emphasis of safety vs. economics can be readily balanced in formulating the design goals.

DESIGN STRATEGIES AND DESIGN PROCESS

The importance of sound management in the design and construction of nuclear power plants has been

stressed by the Committee on Future Nuclear Power Development,¹¹ as shown in the following excerpt:

One of the most important factors affecting the future of nuclear power in the United States is its cost in relation to alternatives and the recovery of these capital and operating charges through rates that are charged for the electricity produced. The industry must develop better methods for managing the design and construction of nuclear plants. Arrangements among the participants that would assure timely, economical, and high-quality construction of new nuclear plants will be prerequisites to an adequate degree of assurance of capital cost recovery from state regulatory authorities in advance of construction.

The design strategies and approaches practiced in each proposed new design can be used for evaluating the adequacy of various conceptual reactor designs. To achieve the buyer's design goals (Table 1), a set of key design strategies for vendors to consider is suggested in Fig. 1. If these design strategies are accepted by the vendor, then the burden of proof that these strategies have been in the design is the responsibility of the vendor. Note that the interrelationships between the strategies and goals are shown by arrows in Fig. 1. The horizontal arrows indicate the strategy effort to achieve the design goal. The vertical arrows indicate the interrelation of one goal to another. For example, the safety goal affects availability, which, in turn, affects the plant economy. The contents of the suggested design strategies are further elaborated in the following text.

Table 1 Design Goals

Safety and reliability	
• Probability of need of emergency evacuation	<10 ⁻⁶ /RY
• Core damage frequency	<10 ⁻⁵ /RY
• Containment rupture during severe accident	<10 ⁻²
• Plant personnel exposure	<100 man-rem/R
Operability and availability	
• Life average availability	>85%
• Advanced control and protection systems, daily load cycling	
• Refueling cycle	18 to 24 months
• Plant life	40 to 60 years
Constructibility and maintainability	
• Standardized design and construction	
• Simplified systems	
• Integrated NSSS/BOP design	
• Planned maintenance to ensure successful operation in entire plant life	
Cost reduction	
• Energy cost and initial capital cost reduced by at least 15% from that of the existing similar plant	
• Construction time from first structure concrete to fuel load: 48 to 66 months	

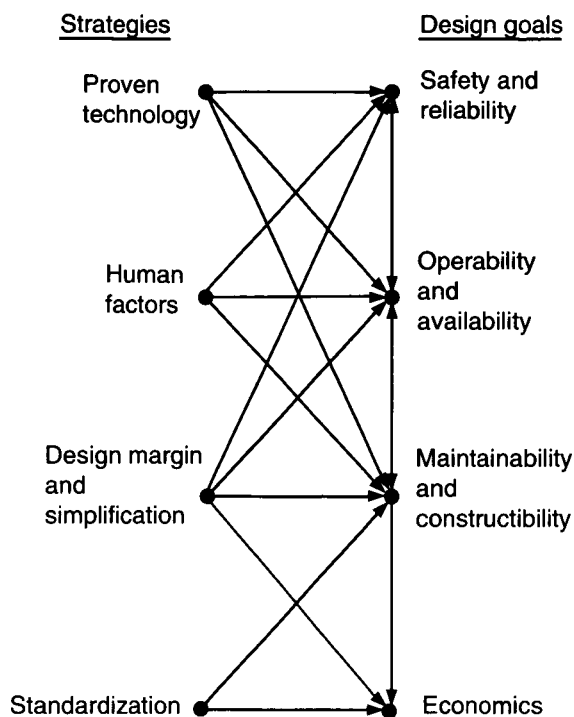


Fig. 1 Key strategies for achieving design goals.

Proven Technology

Proven technology is defined as structures, systems, components, design analysis, and construction techniques that have the same characteristics and materials, working conditions, and environments as those which have been successfully demonstrated in light-water-reactor (LWR) operations or with prototypes that have been successfully tested. The plant designer must review existing data bases of LWR operating experience to identify both positive experience and causes of significant failures. The successfully proven technology should be used throughout the plant to minimize the risk to the plant owner. This is the most important strategy for ensuring that the design goals will be met.

Human Factors

The human factor considerations should be included in the design of reactor systems, facilities, and equipment in a systematic manner. All aspects of plant design for which there is an interface with plant personnel will incorporate such human factor considerations as

- Transparency of plant design to operating staff.
- Monitoring, controlling, and protection functions assigned to plant operators.

- Monitoring and diagnostic functions performed by plant engineers and managers during normal, upset, and emergency conditions.

- Inspection, on-line and off-line surveillance testing, preventive maintenance, and corrective maintenance functions assigned to maintenance personnel.

Man-machine interface systems will use modern digital technology to simplify communication systems. In particular, the main control room will use an advanced control concept in which integrated displays, alarms, procedures, and controls are available to the operators at a compact workstation but without overloading the operators.

Design Margin and Simplification

Significant design margins will benefit the plant by being forgiving and rugged in the following ways:

- Providing designed-in capability to accommodate anticipated transients without causing initiation of engineered safety systems.

- Providing the operator sufficient time to assess and deal with upset conditions with minimum potential for damage.

- Providing margin to enhance system and component reliability and to minimize the potential of exceeding limits (e.g., Technical Specifications) that might require derating or shutdown.

- Providing additional assurance that the longer plant life requirement can be met.

- Providing designed-in capability to accommodate single failures of systems without ensuing release of radioactive materials.

- Providing assurance of feedwater availability.

- Ensuring the designed-in accesses for in-service inspection and in-line component testing.

Areas in which design margin can be emphasized include core thermal margin, reactor coolant system (RCS) hot-leg temperature, coolant inventory, and provision for ensuring availability of a-c power. More design margins in all these areas help to ensure higher plant availability.

With significant design margins, some rapid-response functions can be relaxed and some complicated systems can be simplified. Simplifications in design are also closely associated with standardization. The methods of achieving simplification in design can be categorized into four groups:

1. Adoption of reliable passive safety features in lieu of complicated, active safety features.

a strong voice in the review meetings on new reactor design and construction. Further, this department will be responsible for providing for feedback on previous operational experiences with design and construction in forming the design criteria for the systems, components, and operating instrumentation.

The integrated planning of plant standardization will permit more efficient use of engineering resources for design, procurement, construction, maintenance, and operations. Plant standardization starts with deciding on the software and then proceeds with designing and fabricating the hardware. Software includes licensing and owner's requirements, design criteria, analytical tools and design methods, conceptual design and plant arrangement, equipment-material specifications, and plant Technical Specifications. Hardware includes core and system design; protection, control, and instrument design; and component design. These steps should be carried out in an orderly manner, as shown in Fig. 2 of the design procedure.

The equipment-material specification is the key step for standardization. It determines the cost and quality of downstream engineering processes, such as hardware reference design, procurement, fabrication, installation, and maintenance. A modification of the issued equipment-material specification caused by any change of upstream design software would have a strong impact on plant cost and quality and also on the benefits of standardization. Therefore these specifications and the associated drawings must be carefully planned and reviewed before issuance. The principle of using proven technology should also be observed. The word "proven" means that any piece of equipment or system to be selected in a standardized design, including material selections, must have been successfully proven in the operation of similar plants or must have been sufficiently tested to prove its reliability and safety.

The overall design processes must be reviewed in view of the whole plant. Safety and other design margins of all systems must be evaluated for their integrated effect on the entire plant. A plant can neither be designed system-by-system individually nor discipline-by-discipline individually because many important interactions exist between related systems or disciplines. A design change may be beneficial to one of the individual systems (or disciplines) but detrimental to others. Thus integrated planning and review of the design processes are necessary before introducing any design improvement into a plant that is expected to be standardized. This overall safety and design review serves also as the focus of feedbacks in the design process.

Standardization in design of the next generation of plants can provide a more stable licensing environment to give the buyer better assurance of plant safety, reliability, systems performance, component quality, reliable delivery, maintainability, and available spare parts supply.

EVALUATION OF NEW DESIGN FEATURES

To help the buyer in evaluating the proposals offered by various vendors, a two-step approach is suggested: (1) plant performance analyses are requested from vendors to confirm the designed performance of their proposed plants and (2) a collection of new design features proposed by various vendors is tabulated and categorized according to system functions for evaluation of the maturity in the development of these design features.

When proposing a new plant design, the plant supplier must have a full justification for employing the new design features in the proposed plant. These justifications are usually presented in the plant performance analyses along with the system operating data or prototype testing results.

For the reliability and availability of the designed systems and equipment needed to achieve the desired capacity factor and safe operation of a plant, the plant supplier must successfully perform analyses and provide design and maintenance data as follows:

- Perform safety analysis to meet NRC acceptance criteria and perform transient analysis of performance capabilities to respond to operating events to protect the user's investment. During various stages of design and construction, intermediate safety analyses should be performed by feeding in the up-to-date data (as shown in Fig. 2) to ensure that the safety and reliability criteria are maintained at various stages of design and construction.

- Perform analyses, cite experience, and provide maintenance requirements to demonstrate that the operability and maintainability requirements will be met. For this demonstration to be satisfactory, it is necessary to show substantial design and operating margins through service records or through the use of appropriate, realistic testing.

- Perform an evaluation of potential system failures to show that they are tolerable and the system is quickly recoverable during operation and that the frequency of plant trips could be less than or equal to one per year if the plant is properly operated and maintained.

- Provide an availability-reliability analysis report to show that the plant systems, the supporting maintenance

systems, and the recommended spare parts replacements are adequate to meet the specified safety, reliability, and availability goals. The equipment and spare parts should be identified and readily available.

- Provide a detailed quality assurance plan (i.e., formulated in accordance with the requirement of Appendix B to 10 CFR 50) and all other subsidiary standards or their equivalent as appropriate.

- Supply a reliability assurance maintenance program 1 year before the construction is completed.

- Establish an overall construction logic plan during the detail design period. This plan is used as the basis of an integrated plant design-construction schedule. The construction is not to start until the detail design is essentially completed (80 to 90%).

- Demonstrate in a plant layout that the plant equipment is arranged so that adequate access is provided for inspection, maintenance, repair, and replacement.

- Provide design-basis, operating, and maintenance information that is important to successful plant operation over the plant life. This information includes the fundamental design bases of the plant, such as design-basis requirements, calculations, design descriptions, drawings, test procedures, acceptance criteria, and bases for Technical Specification. In addition, extensive operating and maintenance instructions and procedures, as well as equipment drawings, are required to support long-term operation, inspection, testing, maintenance, and repair or replacement. Because the plant designer and manufacturer can best define the maintenance requirements of the equipment, it is the plant supplier's obligation to deliver the long-range maintenance requirements to the owner before the completion of the plant.

For evaluation of maturity in the development of these design features, the new design features proposed by various vendors are tabulated and categorized according to their system functions. A sample format for listing the collected (but not exhaustive) options of new design features is presented in Fig. 4. Because the options qualifying for competition must first satisfy the mandatory requirements in the bid specifications, the buyer must decide which of the proposed options to enter into the list for comparison. A detailed description of each of the listed sample design features is given as follows:

A. Large, strong, and low-leakage steel containment vessel [for pressurized-water reactors (PWRs)]. A large and strong containment vessel with inside containment spray cooling is used. For a double containment, heat removed from the outside of the steel shell should be provided during an accident. The design of containment

systems and instruments should take into account the most severe pressure and temperature environment encountered in severe accident analysis. The containment room should have an "at grade" maintenance hatch. The elevated emergency water storage tank provides water for long-term residual heat removal (RHR).

B. Integrated reinforced concrete containment vessel [for boiling-water reactors (BWRs)]. The design calls for an integrated reinforced concrete containment vessel with a wetwell vent for release of extremely high pressure in a severe accident. Internal containment waterwall cooling transports heat out of the containment. An elevated suppression pool provides gravity-driven cooling to the core with a waterwall to cool the suppression pool.

C. Low-power density core in a large reactor vessel (for PWRs). A low-power density core provides a 15% core thermal margin and a larger reactor vessel to provide a larger downcomer water gap for lowering the embrittlement on the vessel wall. This core uses burnable poison in fuel and gray control rods to flatten the power peaks. Further, it uses a radial reflector and axial blanket to lengthen the vessel life.

D. Low-power density core with electric fine rod motion control (for BWRs). A low-power density core with smaller fuel rods (9×9) is used to provide a 15% core thermal margin. This case uses axially zoned fuel with higher enrichment and less gadolinium absorber in the upper half of the core to flatten core power peaks. Further, it uses electric fine rod motion during normal operation and hydraulic pressure for scram insertion of control rods.

E. Advanced control rooms using digital computer-based technologies (for LWRs). An advanced control room provides effective systems integration by using digital computer-based techniques. In the aircraft industry, a prominent company attributed its success to systems integration, which entails the flawless management of hundreds of separate entities (i.e., people and advanced technologies). All this ensures optimal performance from the hardware and the subsystems. Even the best ideas fail without systems integration. Systems integration is a vital discipline for virtually every sophisticated program.

Digital computer-based technologies provide the means to design systems with high reliability and flexibility while remaining largely free from many of the drift and calibration problems. Digital computer-based systems can test themselves on an essentially continuous basis without requiring operator attention. This reduces the potential that failures will go undetected and lead to challenges to the safety systems. Furthermore, the new computer-based digital technologies make it practical for such systems to identify the source of a malfunction and

Design goals	Influence paths ← Direct inf. ← - - Indirect inf.	Systems	Feature number	Brief description of design features	
Safety and reliability	●	α Containment systems	A	Large, strong, and low-leakage steel containment vessel	PWR
			B	Integrated reinforced concrete containment vessel	BWR
Operability and availability	●	β Reactor core Control systems	C	Low-power density core in a large reactor vessel with burnable poison in fuel and grey control rod to provide 15% core power margin	PWR
			D	Low-power density core with smaller fuel rod and electric fine rod motion control rod to provide 15% core power margin	BWR
			E	Advanced control rooms using digital computer-based technology and multiplexing	LWR
			F	Ring-forged reactor vessel with low initial RT _{NDT} material	PWR
Constructibility and maintainability	●	γ Cooling flow systems and vessel, piping, pumping, and steam generation	G	Safety-grade depressurization system at hot leg	PWR
			H	In-containment emergency water storage tank (IRWST)	PWR
			I	In-vessel variable-speed circulation pumps	BWR
			J	Combined ECCS and RHR in three redundant and independent divisions and automated RCS depressurization	BWR
			K	Large steam generators using 690 alloy tubing	PWR
Cost reduction	●	δ Supporting systems for NSSS	L	Integrated containment spray and shutdown cooling system	PWR
			M	High-pressure shutdown cooling system	PWR
			N	Improved radwaste treatment system	LWR
		ϵ Managing efforts	O	Standardization of software and hardware in plant design and construction	LWR
			P	Training for operators and maintenance personnel and maintenance technology transfer	LWR

Fig. 4 Sample list of new design features.

provide guidance for correction, which reduces staff effort and plant downtime. These technologies should result in a plant that is easier to operate than current plants and therefore one that is inherently safer and more reliable if the necessary redundancy is provided.

Because the advanced control room may require the first-of-a-kind-engineering (FOAKE) effort done for

the final design, a detailed review on this system is suggested; a list of desired features is also suggested for review, as follows:

- Use of proven, up-to-date technology.
- Computer-based control and monitoring system and multiplexing.

- Controlled, systematic design process.
- Flexibility for modification.
- Continuous, automatic self-testing of systems.
- Automatic execution of periodic surveillance tests (manually initiated).
- Design for maintenance.
- Software design verification and validation program.
- Analysis of reliability and the effects of failures.
- System checkout before delivery.

F. *Ring-forged reactor vessel* (for PWRs). The reactor vessel is constructed of ring-forged sections to avoid any welds in high-influence regions (e.g., core) and manufactured with low initial RT_{NDT} material to preclude brittle fracture.

G. *Safety-grade depressurization system* (for PWRs). A safety-grade rapid depressurization system at the hot leg aids adequate core cooling and eliminates the possibility of a postulated high-pressure core-melt ejection accident. However, precautions must be taken for avoiding inadvertent startup of the solid water plant. A larger pressurizer volume is used for eliminating the pressure-operated relief valve (PORV) at the top of the pressurizer and also reduces the possibility of solid water startup.

H. *In-containment emergency water storage tank (IRWST)* (for PWRs). A large in-containment emergency water storage tank (IRWST) supplies water to the high-head safety injection (SI) pump and RHR/containment-spray pump and operates the letdown and RCS depressurization system. An additional accumulator in each train replaces low-pressure injection pumps. A diesel-operated backup seal injection pump is used in station blackout.

I. *In-vessel variable-speed circulation pumps* (for BWRs). A key feature of the advanced boiling-water reactor (ABWR) design is the elimination of the external recirculation loops and the incorporation of in-vessel pumps for reactor coolant recirculation. All large pipe nozzles to the vessel below the top of the active fuel are eliminated. This alone improves reactor safety by eliminating the loss-of-coolant accident (LOCA) of recirculation loops and reducing the pressure drop of coolant recirculation. In-service inspection (ISI) is also reduced because of the incorporation of internal pumps and the elimination of the recirculation piping and nozzles. This also helps to minimize inspection manpower and reduce radiation exposure.

Variable-speed in-vessel pumps provide 11% excess core flow above the rated flow. Daily load following from

100% to 70% to 100% power (in a 14-1-8-1 hour cycle) is easy with the use of core flow-rate adjustment, and no control-rod movement is needed. Through maximum use of the excess flow and slight control-rod adjustment, load following of 100% to 50% to 100% is easily attainable. In addition, the excess flow capacity allows for a spectral shift operation to provide additional burnup with all rods out for increased operational flexibility, extended operation, and reduced fuel cycle costs.

J. *Combined ECCS and RHR in three redundant and independent divisions and automated RCS depressurization* (for BWRs). The emergency core cooling system (ECCS) and RHR are combined in three redundant and independent divisions for suppression pool cooling and low-pressure core cooling. The RCS depressurization function is automated. Reactor core isolation cooling (RCIC) has been upgraded to a safety system and the steam-turbine-driven pump upgraded to a safety grade.

K. *Large-stream generators using 690 alloy tubing* (for PWRs). Large-steam generators (Model-F or equivalent) using 690 alloy tubing are to be operated at a hot-leg temperature less than 600 °F.

L. *Integrated containment spray and shutdown cooling system* (for PWRs). The containment spray system and the shutdown cooling system are integrated, and pumps are interchangeable. Thus backup and higher reliability are provided for both systems.

M. *High-pressure shutdown cooling system* (for PWRs). Shutdown cooling is designed to maintain piping integrity even if pipes are accidentally exposed to full primary-system pressure. This precludes a large interfacing LOCA.

N. *Improved radwaste treatment system* (for LWRs). Improved radwaste treatment can reduce total radwaste volume to 100 drums per year for a PWR and to 200 drums per year for a BWR.

O. *Standardization of software and hardware in plant design and construction* (for LWRs). A plant design and construction schedule will be established in the detail design stage. Plant construction will not start until 80 to 90% of detail design is completed.

P. *Training for operators and maintenance personnel and maintenance technology transfer* (for LWRs). The vendor is obligated to train the plant operators and maintenance personnel to operate, inspect, test, troubleshoot, maintain, and repair the equipment by providing the prototype control room simulator for operator training and

by transferring to the buyer sufficient design data, procedures, and drawings for long-term maintenance. This information should be systematically organized so that it is readily retrievable and useful.

SAFETY PRINCIPLES AND CREDIBLE TOOLS FOR SAFETY ANALYSIS

Safety Principles

The safety of a nuclear power plant is like the strength of a chain—it is only as strong as its weakest link. No system affecting safety is allowed to be less safe than the entire plant safety limit, but on the other hand, unnecessarily overdesigned safety systems are meaningless and counterproductive. Therefore the principles of reactor safety must be complete and balanced. A set of the most complete and comprehensive safety principles has been suggested in INSAG-5/IAEA:¹⁹

The INSAG-3/IAEA, *Basic Safety Principles for Nuclear Power Plants*, should become mandatory, with the predominant features:

- Defense in depth continues to be the fundamental means of insuring the safety of nuclear plants.
- The three fundamental safety tenets continue to be: maintain cooling, control the power level, and confine the radioactive material.

More specific aspects of design should be addressed as follows:

- The concept of plant design should be extended to include the operating and maintenance procedures that are required.
- Design should avoid complexity.
- The plant should be designed to be user-friendly.
- Design should not make safety depend on early operator action.
- The design of the system used for confinement of fission products should take into account the most severe pressure and temperature encountered in severe accident analysis.
- Accidents that would be large contributors to risk should be designed out or should be reduced in probability and/or consequences.
- The plant should be adequately protected by design against sabotage and conventional armed attack.
- Design features should reduce the uncertainty in the results of the Probabilistic Safety Analysis.
- Consideration should be given to passive safety features.

The preceding principles are straightforward and well-balanced. However, some interpretations might be added (for example, defense in depth includes both mitigation and prevention of an accident). The stated principle on containment design indicates that the containment design

should take the primary responsibility for the defense function of mitigation of severe accident consequences. Since a stronger and larger containment system would only slightly increase the initial construction cost, it is a very appropriate principle. Further, it will not interfere with or complicate other plant operating systems. Conversely, the defense function of preventing accidents should be emphasized in the design of fluid and heating systems, where most operational activities occur. The issue of accident prevention vs. consequence mitigation can be resolved with this approach.

Tools for Safety Analysis

For a tool to achieve the desired balance in accident prevention and consequence mitigation, the author proposes that a "living" PRA program^{20,21} be used. This program should be updated periodically to reflect actual current plant conditions so that it can be used for timely decision making (for example, should a component of a system not meet functional requirements, the original failure probabilities and the risk of the plant need to be reevaluated). To maintain the recalculated plant risk below the required limit in the new safety evaluation, it may be necessary to implement a safety improvement, such as an equipment or system modification or replacement.

Furthermore, the "living" PRA, which is to be used for reactor safety evaluation, should be developed through closely coordinated probabilistic and deterministic analyses as an "orchestrated" approach. The probabilistic analysis relies on the results of deterministic studies for its data inputs. The data spread and the degree of accuracy in simulation of the deterministic studies, in turn, depend on the results of probabilistic analyses for identifying their safety significances.

For instance, the deterministic studies provide the data base to define the characteristics of a scenario and its consequences. In a scenario having severe consequences, the confidence level of the deterministic data base must be high enough in predicting the consequences so that the resulting risk is not underestimated. At the same time, the confidence level in the deterministic consequence evaluation could be reduced for a very low probability event. For example, if the confidence level in the consequence evaluation for an event of frequency $\geq 10^{-4}/RY$ is required at 3σ (i.e., the deterministic consequence being evaluated at the $+3\sigma$ limit), then the confidence level could be $\pm 2\sigma$ for an event of frequency $\geq 10^{-5}/RY$ and $< 10^{-4}/RY$ and could be $\pm 1\sigma$ for $\geq 10^{-6}/RY$; $< 10^{-5}/RY$; then the best estimate of the consequence should be used for anything $< 10^{-6}/RY$. In this way, the resulting design

would not be unnecessarily stringent for mitigating some remote events so that an economically balanced new reactor design could be achieved.

In the risk analysis, any credible multiple-failure scenarios and minor human errors of omission and commission also deserve attention according to the lessons learned from the Three Mile Island Nuclear Station Unit 2 (TMI-2) and Chernobyl accidents. The risks of such failures are generally not directly related to reactor operating power levels. Reduction of risk in these cases must be based on the root cause.

On the basis of the preceding approaches, the important assumptions or the expert estimates contained in current PRAs should be either validated by simulated experiments or verified by analytical calculations. Thus a generally acceptable PRA methodology can be established. It may be used to check the safety of a reactor design against the predetermined safety goal and also to guide the practical direction and extent of the defense in depth. These steps provide a sound basis for selecting the safety design of a new nuclear power plant.

ENERGY COST EVALUATION AND GUIDELINES FOR SELECTION

Evaluation of Cost Benefits of New Design Features

The predicted overall plant performance is suggested to be used as the criterion for selecting a new reactor design. Because the energy cost consists of the expenditures of plant construction cost, fuel cost, and operation and maintenance (O&M) cost as well as the risk costs of public safety and plant availability, this criterion can be expressed in the variation of energy cost of the proposed new plant design with respect to that of a base plant. Then the energy cost is evaluated by the total expenditures divided by the net energy production. Thus energy cost can serve as an indicator of the plant performance in safety, operation, maintenance, and construction.

Results of the review of new design features provide the data base for evaluating the predicted plant performance in terms of individual benefits. The evaluated benefits should be checked and compared with two vendor's reports: the Final Safety Analysis Report (FSAR) and the Design Review Report (DRR).

The format of the evaluation matrix of new design features is provided in Fig. 5. The evaluated individual benefits (or disadvantages) are listed in the columns of Fig. 5 as the following performance benefits:

- X = safety risk reduction (%), including reliability effects
- Y = availability increase (%), including reliability effects, operability effects, and maintainability effects on availability
- W = plant cost reduction (%) resulting from fabrication cost and the reduction in construction time
- Z = energy cost reduction (%) resulting from all the preceding effects

The performance benefits X and Y can be quantified by evaluating the inherent benefit of the concept and the engineering benefit of the design. The values are determined on the basis of the design information in FSARs, DRRs, and the EPRI ALWR-URD. The performance benefit W is a measure of the design adequacy, fabrication ability, and appropriate planning in construction. The performance benefit Z is the net benefit in energy cost of the design. Z is used for optimization of plant designs.

The way to quantify Z, the energy cost change (%), is to calculate the performance benefits of X, Y, and W on the average energy cost of a typical nuclear power plant computed on a life-cycle basis. Various conversion ratios between the performance benefits and the energy cost change in percent are listed in Table 2 for the buyer to consider. Brief rationales for these conversion ratios are given in Appendix B. The numerical values of these conversion ratios vary with site locality factors, however, such as public acceptance of nuclear power, load-follow requirement, fuel resources, site geology, and the weather. Thus the actual conversion ratios to be used in cost evaluation must be decided by the individual buyer on the basis of the local conditions of the site.

From the preceding paragraphs, the benefit of safety improvement is clearly related to the equivalent power rate cost. The conversion ratio of 0.1 for converting the risk reduction (%) into the equivalent energy cost in Table 2 was developed on the basis of the plant safety risk *below* the safety goal (or licensing) risk limit. If plant safety risk is above the safety goal risk limit, the conversion ratio would be much greater than the quoted value of 0.1 to cover the needed plant improvement cost. This conversion process readily resolves the choice of safety vs. economy in a reactor design.

To demonstrate the use of Table 2, the reduction of energy cost (%) is evaluated by the following equation:

$$\begin{aligned} \text{Energy cost reduction (\%)} = & 0.1 (\text{risk reduction in \%}) \\ & + 1.36 (\text{availability change in \%}) \\ & + 0.64 (\text{initial cost reduction in \%}) \end{aligned}$$

Design goals	Influence paths ← Direct inf. ← - - Indirect inf.	Systems	Feature numbers involved in evaluation	Evaluated benefit of energy cost influenced by design features					
				X, % - risk reduction	Y, % availability increase		W, % - initial cost reduction		Z, % Net benefit in energy cost reduction
				Safety improvement based on safety reviews and analyses	Improvement based on Q/A plans and reliability analysis	Operability improvement based on advanced control system and technical specifications	Improvement based on maintenance plan and training plans	Benefit from managing effort and fabrication and construction plans	
Safety and reliability	←	α Containment systems	PWR (A) PWR (B)	X_α	$y_{\alpha 1}$	$y_{\alpha 2}$	$y_{\alpha 3}$	w_α	z_α
		β Reactor core Control systems	PWR (C) E (E) BWR (D) E (E)	X_β	$y_{\beta 1}$	$y_{\beta 2}$	$y_{\beta 3}$	w_β	z_β
Operability and availability	←	γ Cooling flow systems and vessel, piping, pumping, and steam generation	PWR (G) H (H) K (K) BWR (I) J (J)	X_γ	$y_{\gamma 1}$	$y_{\gamma 2}$	$y_{\gamma 3}$	w_γ	z_γ
Constructibility and maintainability	←	δ Supporting systems for NSSS	PWR (L) M (M) BWR (N) N (N)	X_δ	$y_{\delta 1}$	$y_{\delta 2}$	$y_{\delta 3}$	w_δ	z_δ
Cost reduction	←	ε Managing efforts	LWR (O) P (P)	X_ϵ	$y_{\epsilon 1}$	$y_{\epsilon 2}$	$y_{\epsilon 3}$	w_ϵ	z_ϵ

Fig. 5 Evaluation matrix of new design features.

The reduction of equivalent energy cost for each of the new reactor features is listed in the last column (Z) of Fig. 5. The net benefit in energy cost for each system can be calculated as:

$$Z_\alpha = 0.1(X_\alpha) + 1.36(Y_{\alpha 1} + Y_{\alpha 2} + Y_{\alpha 3}) + 0.64(W_\alpha)$$

$$Z_\beta = 0.1(X_\beta) + 1.36(Y_{\beta 1} + Y_{\beta 2} + Y_{\beta 3}) + 0.64(W_\beta)$$

$$Z_\gamma = 0.1(X_\gamma) + 1.36(Y_{\gamma 1} + Y_{\gamma 2} + Y_{\gamma 3}) + 0.64(W_\gamma)$$

$$Z_\delta = 0.1(X_\delta) + 1.36(Y_{\delta 1} + Y_{\delta 2} + Y_{\delta 3}) + 0.64(W_\delta)$$

$$Z_\epsilon = 0.1(X_\epsilon) + 1.36(Y_{\epsilon 1} + Y_{\epsilon 2} + Y_{\epsilon 3}) + 0.64(W_\epsilon)$$

Then the net benefit in energy cost for a proposed plant is

$$Z = Z_\alpha + Z_\beta + Z_\gamma + Z_\delta + Z_\epsilon$$

The Z values for respective proposed plants are compared in the selection process.

Table 2 Conversion Ratios

	Equivalent energy cost reduction, %
1% plant cost	0.64
1% fuel cost	0.23
1% operation and maintenance cost	0.13
1% nuclear steam supply system cost	0.05
1% availability factor ^a	1.36
1-year plant life (2.5% plant cost)	1.60
1% total risk reduction (below safety goal risk limit)	0.10
1% worker dose	0.10

^aAll intangible benefits are evaluated in terms of an availability factor.

Guidelines for Selection of New Reactor Designs

As a conclusion to this article, guidelines for selection are as follows:

1. To establish a set of synchronized and balanced design goals (see Table 1) by the potential buyer(s). The emphasis of reactor safety and management policy issues should be properly balanced in formulating the design goals.

2. To review the vendor's design strategies (or criteria) and management approach in the design. The design strategies should be oriented to meet the design goals and to implement the lessons learned from previous operating experiences (see Fig. 1). The management approach should be able to carry out the design strategies effectively and to consolidate the expertise of various disciplines and all responsible departments in an integrated plant design procedure (see Fig. 2).

3. To review the following design requirements of each of the vendors:

- The validity of the reliability analysis and test results, which are used by the vendors to qualify their new design features. All significant features are to be listed in the format of Fig. 4.
- The consistency of the functions and specifications of the designed software and hardware, which are to be standardized throughout the plant.
- The adequacy of the designer's QA plans, testing procedures in the design and fabrication stages, respectively, maintenance plans (short- and long-range) with the spare parts supply information, and the maintenance personnel training plans.

4. To review the control and protection systems, instrumentation and communication systems, operating Technical Specifications, and the operator training programs to ensure that the design goals of safety and operability will be met.

5. To review the designer's safety analysis to ensure the licensability of the plant to operate at a specific site.

6. To record the evaluation results of the above reviews in the format of Fig. 5 and to convert the evaluated influence benefit of the design features into the percentage reduction of energy cost of a proposed plant. The net reduction of energy cost of each proposed plant should be compared. Then the final selection can be made objectively.

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APPENDIX A DESIGN REQUIREMENTS

Table A.1 Key Utility Design Requirements for Advanced Light Water Reactor^{a,b}

Plant size	Reference size 1 200-1 300 MW(e) for evolutionary designs, reference size 600 MW(e) for passive safety designs		
Design life	60 years		
Design philosophy	Simple, rugged, no prototype required		
Accident resistance	≥15% fuel thermal margin, increased time for response to upsets		
Core damage frequency	<10 ⁻⁵ /RY by probabilistic risk analysis		
Loss of coolant accident	No fuel damage for 6 in. pipe break		
Severe accident mitigation	< 25 rem at site boundary for accidents with >10 ⁻⁶ /year cumulative frequency		
Emergency planning zone	For passive plant provide technical basis for simplification of off-site emergency plan		
Design availability	87%		
Refueling interval	24-month capability		
Maneuvering	Daily load follow		
Worker radiation exposure	<100 person-rem/year		
Construction time	1,300 MW(e): ≤54 months (first concrete to commercial operation); 600 MW(e): ≤42 months		
Design status	90% complete at construction initiation		
Economic goals	10% cost advantage over alternatives (nonnuclear) after 10 years and 20% advantage after 30 years		
Resulting cost goals (1989 \$)	1 200 MW(e)	600 MW(e)	
	Overnight capital	1 300 \$/kW(e)	1 475 \$/kW(e)
	30-year levelized	6.3 cents/kWh	7.2 cents/kWh
	total generation	1 200 MW(e) commercial operation in 1998; 600 MW(e) in 2 000	

^aExcerpted from Ref. 11.

^bThese requirements apply to both the large evolutionary LWRs and to the mid-sized LWRs with passive safety features.

APPENDIX B COST EVALUATION

(Excerpted from INPO, *Performance Indicators for the U.S. Nuclear Utility Industry—1992 Year-End Report*, Report INPO 93-003, March 1993).

The evaluation of innovations is essentially a cost-benefit study. In the present evaluation, however, both the cost and the benefit are expressed in a nondimensional form, namely, a percentage of energy

cost (%). An energy cost of 1% is equal to 1% of the cost of energy from a typical nuclear plant computed on a life-cycle basis. Energy cost on a life-cycle basis is calculated using the simplified formula

$$\text{Energy cost} = \frac{\text{Yearly capital cost} + \text{yearly fuel cost} + \text{yearly O \& M cost}}{\text{Capacity factor } 365 \times 24 \times \text{rated power}}$$

The actual values of the yearly capital cost, yearly fuel cost, yearly O&M cost, and capacity factor are

plant-specific and vary over time. Enough general information is available, however, to be useful regarding the relative magnitudes of these components for nondimensional analysis. For example, AIF-INFO 85, February 1985, provides the following information, based on statistical analyses of power cost in the United States. For U.S. nuclear plants the percentage composition of life-cycle power cost is as follows: yearly capital cost (64%), yearly fuel cost (23%), and yearly O&M cost (13%). These numbers vary with time and locale.

DOE Report DOE/NE-0044/2 of 1984 allows breaking the plant capital cost down into its major components as follows: NSSS equipment (8%), turbine generator equipment (6%), BOP equipment (14%), construction (13%), allowance for escalation (28%), and allowance for interest (31%). The total percentage is 100%. Note that both escalation and interest are directly affected by the length of the construction time and the current interest rate.

These data can now be used to establish conversion ratios to determine the cost-benefit ratio of various plant changes. Of course, these values are approximate and represent the author's tentative values to show the relative importance of various innovations and to demonstrate the proposed evaluation method. It is recognized that final conversion ratios have to come from actual design studies using economic parameters selected to match a particular plant.

Impact of improved reliability. The improvement in plant reliability can be evaluated through the impact on the plant availability factor. Plant changes that increase the plant availability factor, through improved plant availability or reduction in the forced outage rate, will reduce the percentage of energy cost. Because the percentage of energy cost is inversely proportional to the plant availability factor and typical U.S. plant availability factors (INPO, *Performance Indicators for the U.S. Nuclear Utility Industry—1992 Year-End Report*, Report INPO 93-003, March 1993) are around 0.737, a 1% change in availability factor will decrease life-cycle power cost by a factor of $1/0.737 = 1.36$. Hence, a 1% change in availability factor is worth about 1.36% energy cost.

Impact of reduced fuel cost. The weight of fuel cost in the life-cycle power cost is 23%. Hence, a 1% change in the yearly fuel cycle cost is equivalent to a reduction of 0.23% energy cost.

Impact of plant equipment cost. This impact is demonstrated by using the cost of NSSS equipment. The weight of the NSSS equipment cost is 8% of the plant's initial capital cost. Thus a 20% change in NSSS cost is equivalent to a $(20 \times 0.08 \times 0.64 = 1.02)$ % change in energy cost. In other words, or, a 1% change in NSSS equipment cost is equal to 0.05% energy cost.

Impact of plant life. Based on a 40-year plant life, a year of prolonged plant life is equivalent to 2.5% of plant cost (neglecting the plant renovation cost). Thus, to increase 1 year of plant life is equivalent to $2.5\% \times 0.64 = 1.6\%$ energy cost.

Impact of O&M cost. The weighting for operation and maintenance cost is 13% of the life-cycle power cost. Thus, a 10% change in O&M cost is equivalent to 1.3% energy cost. Note that the increase in O&M cost could improve availability or forced outage rate. This benefit is accounted for by increasing the availability factor.

The beneficial effect on public acceptance of those changes that are related to occupational radiation exposure at the plant and the total safety risk of the plant can be converted into equivalent values of the percentage of energy cost through the use of engineering judgment. For the analysis in this article, it is assumed that a reduction of 0.1% energy cost is equivalent to a 1% further reduction in the total risk below the current limit in a plant safety goal. A reduction of 0.1% energy cost is equivalent to a 1% reduction in occupational radiation exposure.

There are many intangible benefits from design improvements, e.g., enhancement of plant reliability, public acceptance, ease of maintenance, flexibility of operation, etc., that cannot be directly put into dollar figures. It might be possible, however, to convert them into a nondimensional form of some tangible features, such as an availability factor. Naturally, these conversion ratios would involve some subjectiveness. It is the owner's prerogative to select the conversion ratios for use in his decision-making process. The conversion ratios in Table B.1 are used only for examples. The buyer selected conversion ratios should be used as a basis for evaluation of design improvements of PWRs and BWRs.

It should be noted that the incremental cost changes given here include only the equipment and its design cost, but they do not include the costs of the first-time design and the associated research and development (R&D).

Table B.1 Conversion Ratios

	Equivalent energy cost reduction, %
1% plant cost	0.64
1% fuel cost	0.23
1% O&M cost	0.13
1% NSSS cost	0.05
1% availability factor ^a	1.36
1-year plant life (2.5% plant cost)	1.60
1% total risk reduction (below safety goal risk limit)	0.10
1% worker dose	0.10

^aAll intangible benefits are evaluated in terms of an availability factor.

Review of Nuclear Piping Seismic Design Requirements

By G. C. Slagis^a and S. E. Moore^b

Abstract: *Modern-day nuclear plant piping systems are designed with a large number of seismic supports and snubbers that may be detrimental to plant reliability. Experimental tests have demonstrated the inherent ruggedness of ductile steel piping for seismic loading. Present methods to predict seismic loads on piping are based on linear-elastic analysis methods with low damping. These methods overpredict the seismic response of ductile steel pipe. Section III of the ASME Boiler and Pressure Vessel Code stresses limits for piping systems that are based on considerations of static loads and hence are overly conservative. Appropriate stress limits for seismic loads on piping should be incorporated into the code to allow more flexible piping designs. The existing requirements and methods for seismic design of piping systems, including inherent conservatism, are explained to provide a technical foundation for modifications to those requirements.*

Piping system design changed significantly in the early 1970s. Seismic requirements, rather than design for thermal expansion effects, became the focus. This change of focus resulted from the emphasis on seismic hazards through the licensing process and publication, by the Nuclear Regulatory Commission (NRC), of regulatory guides and the standard review plan. The regulatory decision to require linear-elastic response spectra methods with low damping was made at a time when building spectra were relatively benign and analytical methods were at an early stage of development. Since that time continual modification of the methods for implementation of linear-elastic dynamic analysis has occurred. Many technical issues, such as number of modes, rigid mode effects, support stiffness, and building amplification of ground motion, have been resolved by more rigorous analytical techniques. The engineering and design costs, however, have increased dramatically as a result of these changes.

Seismic loading is the dominant consideration for selection of supports for a piping system. Piping is

typically supported so that the first natural frequency is above the peak of the response spectra to keep pipe seismic stresses within the *American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code* Sec. III allowables. This results in a large number of lateral supports. Many of these supports must be snubbers to allow flexibility for thermal expansion growth. The supports are relatively massive to resolve technical concerns on support stiffness effects. The number and size of supports lead to complex support designs and substantial civil structures. The resulting congestion significantly impacts construction and maintenance of all plant equipment. Maintenance of snubbers is a major effort. In-service examination and tests of snubbers are required by Sec. XI of the *ASME Boiler and Pressure Vessel Code* to verify that the snubbers will function properly.

The appropriateness of the present methods for seismic design of piping systems has been a concern within the nuclear industry since the early 1980s. Many analytical and experimental studies have been conducted on dynamic response of piping systems to seismic events. These studies indicate that piping has an inherent capability to withstand strong earthquake motions without damage.

A flexible piping system—a system with a low fundamental frequency—is a simpler design. There are fewer supports and snubbers on the pipe, and the support designs can be less complicated. Reverting to flexible systems in comparison with the modern-day “rigid” designs may enhance safety and reliability while reducing construction costs and schedules. With fewer supports that could malfunction, the reliability of the system should be enhanced. Seismic supports adversely affect the thermal expansion characteristics of a piping system. With fewer seismic supports, the safety of the system for normal operating conditions of thermal expansion cycling should be enhanced.

This article provides a technical basis for understanding the present methods used for seismic design of nuclear piping systems and the inherent conservatism. Descriptions of the federal requirements and the Sec. III requirements are provided. Present seismic stress criteria are delineated. The use of two earthquakes—an operating

^aG. C. Slagis Associates, Walnut Creek, CA.

^bOak Ridge National Laboratory, Oak Ridge, TN; managed by Martin Marietta Energy Systems, Inc., for the U.S. Department of Energy under Contract No. DE-AC05-84OR21400.

basis earthquake (OBE) and a safe shutdown earthquake (SSE)—in the design of piping is discussed. Conservatisms in the code rules are summarized. Section III does not specify analytical methods to be used to predict seismic loads. A brief history of the development of linear-elastic response spectra methods is given. Piping has been tested to seismic levels much greater than allowed by the Sec. III stress limits without failure. An overview of the experimental results is provided.

FEDERAL REQUIREMENTS

The *Code of Federal Regulations* (CFR), Title 10, Part 50,¹ provides the regulations for the licensing of nuclear production and utilization facilities. Part 50.55a establishes the requirements for codes and standards. The general requirement follows:

Structures, systems, and components shall be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

Piping and other components that are a part of the reactor coolant pressure boundary must meet the requirements for Class 1 components in Sec. III of the *ASME Boiler and Pressure Vessel Code*.² There are two exceptions:

- (i) In the event of postulated failure of the component during normal reactor operation, the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system; or
- (ii) The component is or can be isolated from the reactor coolant system by two valves in series (both closed, both open, or one closed and the other open). Each open valve must be capable of automatic actuation, and assuming the other valve is open, its closure time must be such that, in the event of postulated failure of the component during normal reactor operation, each valve remains operable and the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only.

Quality Group B piping must meet the requirements for Class 2 components in Sec. III, and Quality Group C piping must meet the requirements for Class 3 components. Guidance for quality group classifications is given in *Regulatory Guide 1.26*³ and Sec. 3.2.2 of the *Standard Review Plan*⁴ (SRP). Industry documents, such as American National Standards Institute and American Nuclear Society publications, also discuss quality group classifications.

Appendix A of 10 CFR 50 is *General Design Criteria for Nuclear Power Plants*; 64 criteria are given to establish the minimum requirements for a nuclear power plant. Criterion 2, *Design Basis for Protection Against Natural Phenomena*, states the following:

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.

Part 100 of 10 CFR is *Reactor Site Criteria*. Appendix A of 10 CFR Part 100 is *Seismic and Geologic Siting Criteria for Nuclear Power Plants*. In this appendix, SSE and operating basis earthquake (OBE) are defined. Required investigations are specified, and the basis for determining the SSE is given. A minimum SSE of 0.1 g is required. The maximum vibratory ground acceleration of OBE must be at least one-half the maximum vibratory ground acceleration of SSE.

SECTION III REQUIREMENTS

Section III provides the rules for design, construction, stamping, and overpressure protection for nuclear power-plant pressure-retaining components, such as piping and vessels. Rules for supports for pressure-retaining components are also provided. Requirements for quality assurance, materials, fabrication, installation, examination, and testing are given. These requirements vary with the classification of the piping. Design requirements for Class 1 piping are given in Subsection NB-3600; Class 2 piping, in NC-3600; and Class 3 piping, in ND-3600.

The design criteria of Sec. III provide stress limits that protect the structural integrity of the pressure boundary. However, requirements on analysis methods to predict seismic loads are not included in the scope of Sec. III. Definition of acceptable analytical methods is a regulatory function. The code establishes limits for Design Loadings, Service Loadings (Levels A, B, C, and D), and Test Loadings. Required loadings and load combinations are not included in the scope of Sec. III. The owner is responsible for identifying the loadings and combinations of loadings in the Design Specification for the component.

A certified Design Specification must be provided for every Sec. III component. The Design Specification must contain sufficient detail to provide a complete basis for construction. The Design Specification must be consistent with regulatory requirements as given in regulatory guides and the SRP. Section 3.9.3 of the SRP establishes OBE as a Service Level B loading and SSE as a Service

Level D Loading. OBE is to be combined with sustained loads and system operating transients. SSE is to be combined with either (1) sustained loads plus loss-of-coolant accidents (LOCA) or (2) sustained loads plus either design-basis pipe breaks or main steam and feedwater pipe breaks.

The Code does not address the issue of operability or function. If function during or after the earthquake is a concern, then the Design Specification must provide the corresponding requirements.

The N-certificate holder for the piping system is responsible for the achievement of structural integrity. The N-certificate holder must also provide a certified Design Report that demonstrates that the as-constructed piping system meets all the requirements of the Design Specification and Sec. III. After satisfactory completion of required examinations, tests, and inspections, and with the authorization of the Authorized Nuclear Inspector, the N-certificate holder applies the N symbol stamp to the piping system. This certifies that the piping system is constructed according to the requirements of Sec. III.

STRESS CRITERIA

Stresses are categorized as primary, secondary, or peak stresses in the Sec. III rules, and the acceptance criteria for each category are different. Primary stresses are developed by an imposed mechanical loading and are necessary to satisfy equilibrium. Primary stresses are load-controlled and are not self-limiting. Hoop stress in straight pipe from pressure is an example of a primary membrane stress. The nominal bending stress in straight pipe from weight is an example of a primary bending stress.

Secondary stresses are developed by the constraint of adjacent material or by self-constraint of the structure. By definition, a secondary stress is self-limiting. The range of secondary stress is controlled to ensure that the piping system shakes down to elastic action. Stress in a piping system as a result of constraint of free thermal expansion is an example of secondary stress. A peak stress is a result of local discontinuities or local thermal stress, including the effects, if any, of stress concentrations. Cyclic peak stresses are evaluated to prevent a fatigue failure.

Summaries of the Class 1 and Class 2 or 3 piping design requirements are given in Figs. 1 and 2. Note that the fatigue evaluation is performed only for Level A and Level B conditions. Primary stresses are evaluated for Design, Levels B, C, and D loadings. The limits for primary stresses increase from Design to Level B to Level C and to Level D. Level D limits are 100% higher than those for Design. The basis for these increases is that

a reduced factor of safety is acceptable for the lower probability of loading.

For Level B (upset conditions) Service Limits, the loadings are expected to occur during the life of the component. The piping and supports must withstand these loadings without damage requiring repair. Level C Service Limits permit large deformations in areas of structural discontinuity that may necessitate the removal of the component from service for inspection or repair. Level C events (emergency conditions) have a low probability of occurrence. Level D Service Limits permit gross general deformations with some consequent loss of dimensional stability and damage requiring repair. The piping may have to be removed from service if the Level D service limits are approached. Level D events (faulted conditions) are extremely low-probability postulated events.

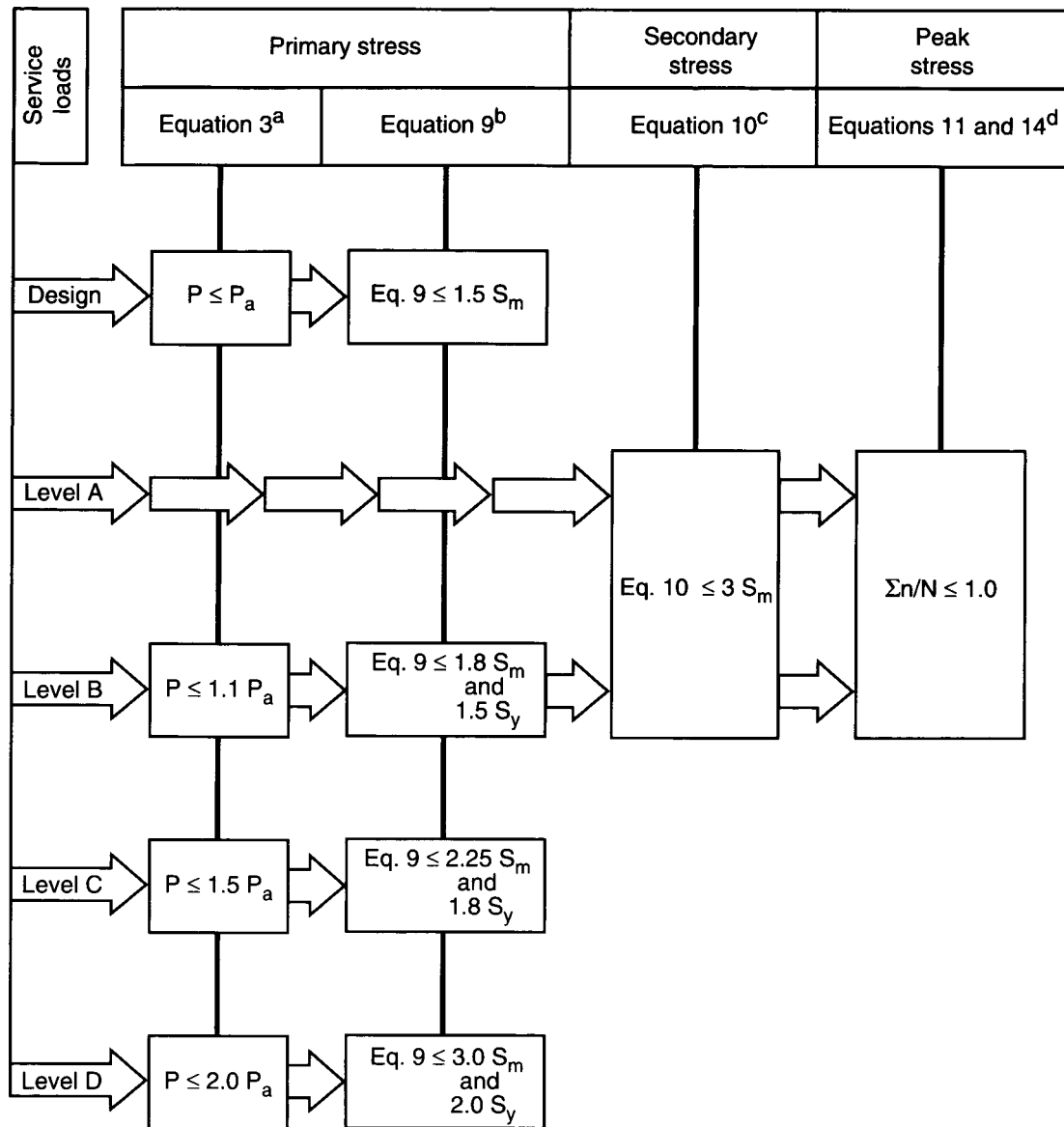
SEISMIC STRESS LIMITS

Section III does not specify the loads or load combinations to be evaluated under Design, Service, or Test Limits. The owner establishes the loads according to regulatory requirements. The Design Specification must define the number of earthquake events and the number of cycles per event. On the basis of the SRP, the OBE is a Service Level B load and the SSE is a Service Level D load. At least one SSE and five OBEs should be assumed. Appendix N of Sec. III suggests ten equivalent maximum stress cycles per event as an appropriate number. The SRP states that the number of cycles per earthquake should be obtained from the synthetic time history (with a minimum duration of 10 seconds) used for the system analysis, or a minimum of ten maximum stress cycles per earthquake may be assumed.

For seismic loadings, two different effects are considered. Inertial loads resulting from the acceleration of the pipe are considered as "load-controlled." Effects of seismic anchor motions (SAMs) are considered as "displacement-controlled." Therefore stresses in the piping system from inertial loads are primary stresses; stresses from SAM are categorized as secondary and peak stresses.

Evaluation of seismic loading for Class 1 piping is illustrated in Fig. 3. The moment resultant (one-half range) from the OBE inertial loads is included in Level B (Eq. 9) [NB-3652]^a with other mechanical loads. The allowable is $1.8 S_m$ or $1.5 S_y$, whichever is lower [NB-3654]. The moment resultant from OBE inertial and

^aSpecific paragraphs of the ASME Boiler and Pressure Vessel Code are referred to by the Code's numbering scheme.



^a Equation 3 predicts the design limit allowable internal pressure, P_a , for satisfaction of primary stress limits.

^b Equation 9 predicts the primary membrane plus bending stress intensity.

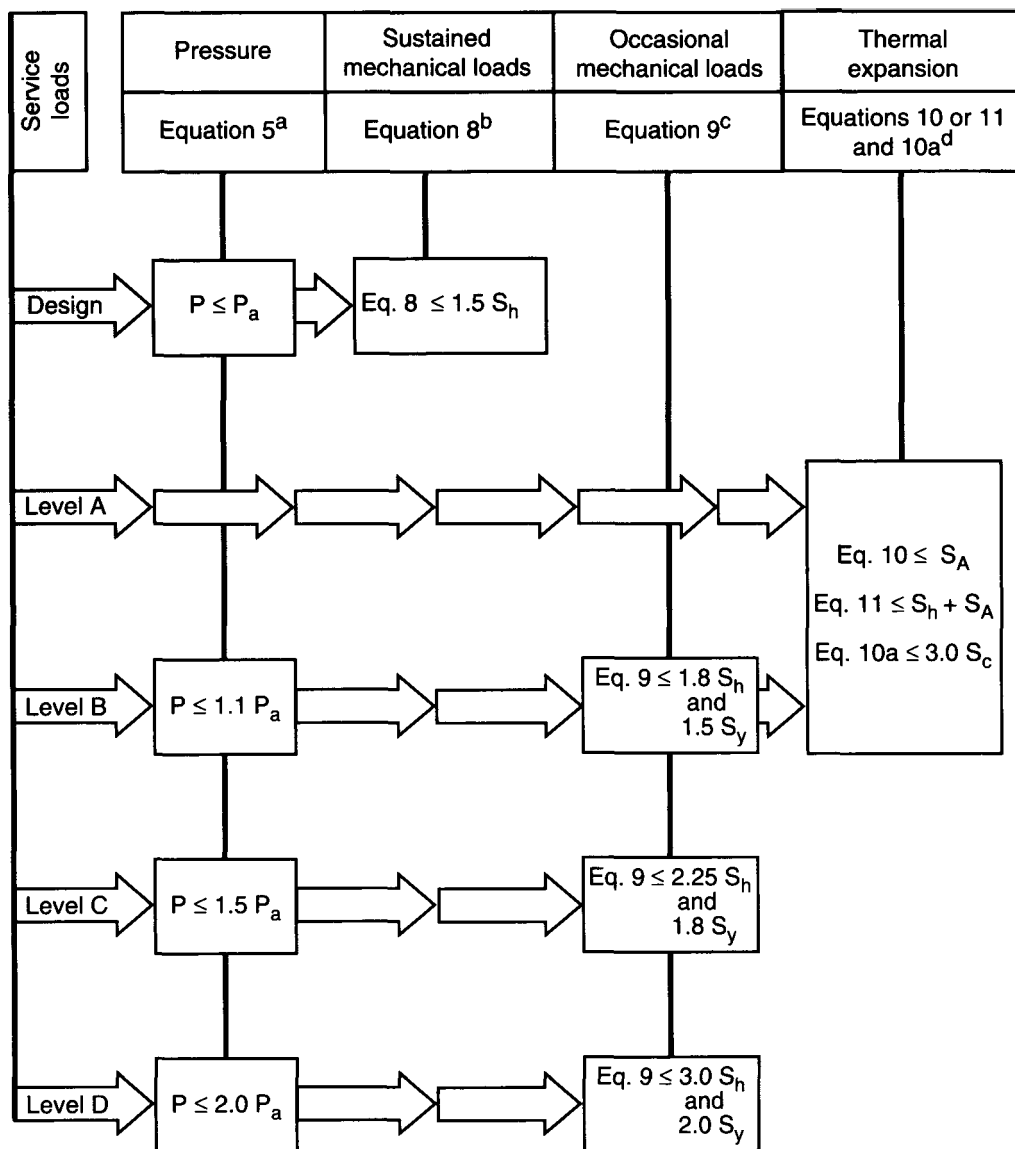
^c Equation 10 predicts the primary plus secondary stress intensity range.

^d Equations 11 and 14 predict the peak stress intensity range for calculation of the cumulative damage. n is the number of cycles of loading; N is the allowable number of cycles from the fatigue curve.

Fig. 1 Class 1 piping design requirements.

anchor motions is included in Eqs. 10 and 11 [NB-3653] for the fatigue evaluation for Levels A and B. Fatigue damage is calculated for the full range of earthquake

stresses. For the SSE event, one-half the range of the seismic inertial load is included in Level D (Eq. 9) with other mechanical loads. The allowable is $3.0 S_m$ or $2.0 S_y$



^a Equation 5 predicts the allowable working pressure, P_a , for design limits.

^b Equation 8 predicts the sustained mechanical load stresses for design conditions.

^c Equation 9 predicts the stresses from sustained and occasional mechanical loads.

^d Equations 10 and 11 predict thermal expansion stresses. Equation 10a predicts stresses from any single nonrepeated anchor movement.

Fig. 2 Class 2 or 3 piping design requirements

[NB-3656]. SAM loads for the SSE earthquake are not evaluated because only primary stresses are controlled for Level D.

Seismic considerations for Class 2 and 3 piping are not as straightforward as those for Class 1 because of the lack of a detailed fatigue analysis methodology in the

Code. Figure 4 illustrates the piping seismic requirements for Class 2 and 3 piping. Earthquake inertial loads are considered as occasional loads. One-half the range of the OBE moment resultant is included in Level B (Eq. 9) [NC-3653] with other sustained or occasional loads. The allowable is $1.8 S_h$ or $1.5 S_y$, whichever is lower.

	M-Inertia 1/2 range	M-Inertia + M-SAM full range
Level B OBE	Equation 9 $1.8 S_m$ and $1.5 S_y$	Equations 10 and 11 $\Sigma n/N \leq 1.0$
Level D SSE	Equation 9 $3.0 S_m$ and $2.0 S_y$	Not evaluated

Fig. 3 Class 1 piping seismic evaluation. Note: SAM is seismic anchor motion, OBE is operating basis earthquake, and SSE is safe shutdown earthquake.

	M-Inertia 1/2 range	M-SAM 1/2 range
Level B OBE	Equation 9 $1.8 S_h$ and $1.5 S_y$	Include in Equation 9 or in Equation 10*
Level D SSE	Equation 9 $3.0 S_h$ and $2.0 S_y$	Not evaluated

*If the SAM moment amplitude is included in Equation 10, it is combined with the range of thermal expansion loading.

Fig. 4 Class 2 and Class 3 piping seismic evaluation. Note: SAM is seismic anchor motion, OBE is operating basis earthquake, and SSE is safe shutdown earthquake.

A rigorous fatigue analysis, as in Class 1, is not required for OBE seismic effects. The analyst has two choices: he can include one-half the SAM moment resultant in Eq. 9 for occasional loads along with the inertial effects, or he can include the SAM moment resultant in Eq. 10 [NC-3653] with the thermal expansion moment range without the inertial effects. An official ASME Code interpretation⁵ specifies that only one-half the SAM range need be included with the thermal expansion moment. Equation 10 is a fatigue-based evaluation and implicitly assumes a minimum of 7000 cycles.

For the SSE event, one-half the range of the seismic inertial load is included in Level D (Eq. 9) with other mechanical loads. The allowable is $3.0 S_h$ or $2.0 S_y$ [NC-3655]. Seismic anchor loads for the SSE earthquake are not evaluated because only primary stresses are controlled for Level D.

TWO-EARTHQUAKE APPROACH

The existing philosophy of seismic design, the concept of two levels of earthquake, goes back to the 1960s and

the recommendations of Newmark and Hall.^{6,7} The design earthquake, or OBE, is of the magnitude that could be expected to occur during the life of a power plant. Therefore the structures and equipment should be *designed* to withstand this earthquake with conventional allowables. The maximum credible earthquake, or SSE, is an extremely low-probability event, but the structures and equipment necessary to prevent a nuclear incident must *survive* this event to ensure safety. The acceptance criterion for the SSE was not specified by Newmark. The allowables would be higher than the conventional allowables used for design and would be based on a failure criterion.

For piping, the acceptance criteria given in Sec. III are consistent with the Newmark and Hall earthquake approach. OBE is classified as a Level B load. The Level B requirements are such that the piping and supports are *designed* to withstand these loadings without damage, including fatigue effects. SSE is classified as a Level D load. Level D Service limits permit gross deformations with some consequent loss of dimensional stability and damage requiring repair or replacement. But the Level D

service limits ensure that the piping will *survive* the SSE event without loss of pressure integrity.

CONSERVATISMS IN CODE RULES

Piping stress limits, as given in NB-3600, NC-3600, and ND-3600 of Sec. III, are based on the use of linear-elastic analysis methods to predict the loads in the piping system. A detailed discussion of the evolution of the code rules for piping design and an explanation of the technical basis for the Levels C and D primary stress limits are given by Slagis.⁸ From study of the development of the Code rules, it is apparent that the primary stress limits are based on static loading considerations. The allowables for static, monotonically applied loads are the same as they are for dynamically reversing loads such as seismic loads. It is reasonable to state that the Code limits are conservative for seismic loads. More appropriate stress limits for piping seismic loads should be incorporated into the Code.

The regulatory requirement of designing for five OBEs where OBE is at least one-half SSE is extremely conservative. The probability of a one-half SSE occurring in a 40-year plant life is extremely low, let alone five of them. Most piping designs are controlled by the OBE requirement (i.e., seismic supports are added to the system to meet the Level B stress limits. Defining the OBE magnitude at a level that is *expected* to occur in a 40-year plant life would result in piping systems with fewer supports.

PREDICTIONS OF SEISMIC LOADS

Section III provides stress limits for evaluation of seismic loads in piping systems, but the Code does not

specify the analytical methods for prediction of the loads. Definitions of acceptable analytical methods to calculate seismic loads in piping systems are given by the SRP and regulatory guides.

A *Nuclear Safety* article by Moore in 1962 (Ref. 9) described seismic considerations in nuclear plants. Developments in analytical methods from the 1960s to the present have been described by Slagis.⁸ Early nuclear structures were designed to a lateral force requirement of the Uniform Building Code. A seismic coefficient of 0.1 to 0.2 g was typical. An Atomic Energy Commission report (TID-7024) describes calculations of seismic stresses based on approximate hand calculations, and seismic loads of 0.5 g were considered unusually high.¹⁰

In the licensing of San Onofre Unit 1, a response spectra approach, based on a Housner spectra, was used for design. The peak of the spectra (1 g horizontal) was used to design critical piping. In the late 1960s, a factor on the design spectra was included to account for building amplification. Not until the early 1970s were building response analyses used to predict floor response spectra for design of piping. Also, it was not until the early 1970s that production computer programs were available for dynamic analyses of piping systems.

The present methods for prediction of seismic loads on piping systems were established by NRC in the early 1970s. The SRP, published in 1975, requires linear-elastic analysis methods with low damping for all nuclear power-plant seismic category structures and equipment. A list of applicable regulatory guides is given in Table 1. An overview of the seismic design process for piping is given in Fig. 5.

With the publication of the SRP, seismic design of piping systems became design by analysis and, specifically, linear-elastic dynamic analysis. The "improvements"

Table 1 Regulatory Guides on Seismic Design

Guide		Year
1.29	Seismic Design Classification	1972
1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants	1973
1.61	Damping Values for Seismic Design of Nuclear Power Plants	1973
1.92	Combining Modal Responses and Spatial Components in Seismic Response Analysis	1973
1.122	Development of Floor Response Spectra for Seismic Design of Floor-Supported Equipment or Components	1976

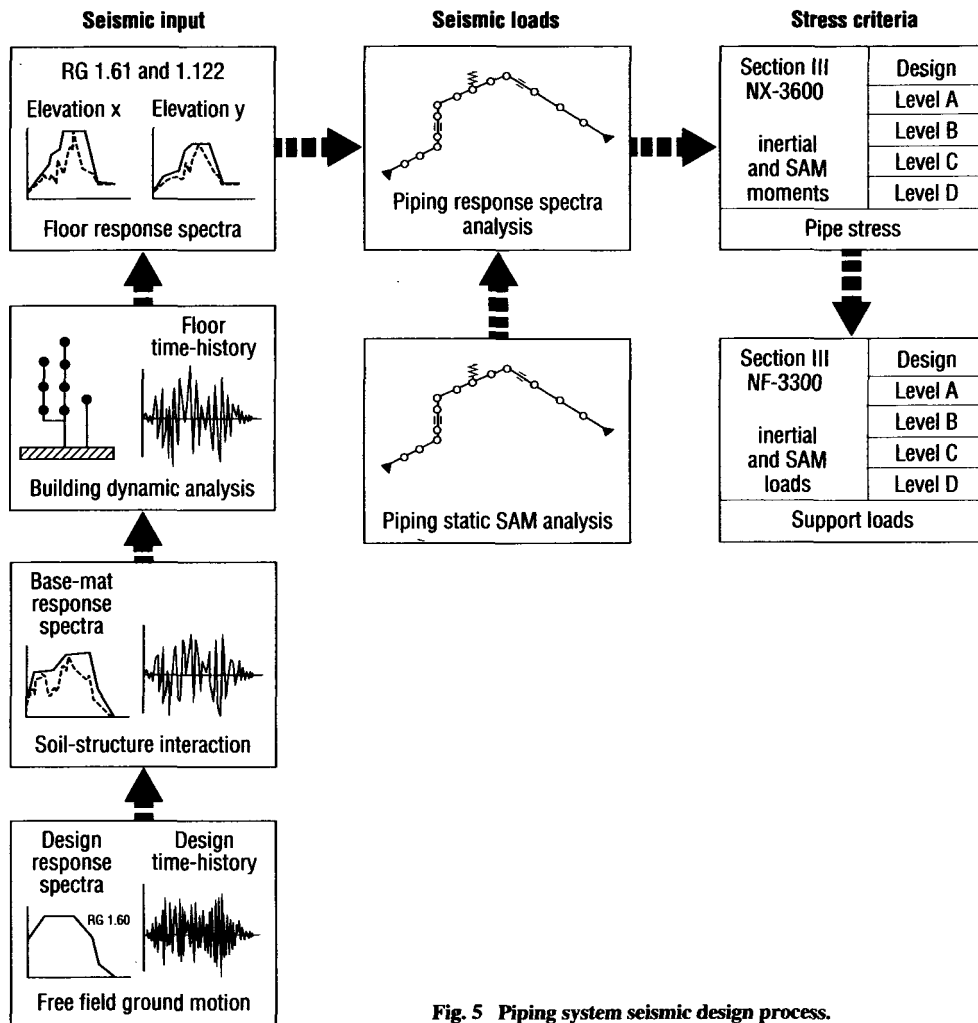


Fig. 5 Piping system seismic design process.

in the technology after 1975 became refinements in the analytical techniques rather than better design approaches. Number of modes considered, proper modal combinations, number of mass points, high-frequency (missing mass) effects, multiple input response spectra, influence of support stiffness, effects of gaps—all these issues on analysis techniques occupied the attention of the industry rather than proper design. Lin¹¹ gives an overview of recent technical considerations on seismic analysis of piping systems.

At the same time that piping analysis techniques were being refined, techniques for definition of the response spectra for design of the piping were being refined. Each step in the process was scrutinized, including definition of the ground motion, soil-structure interaction, modeling of buildings, and floor flexibility effects. Typically,

each step in the analytical process was refined on the basis of philosophy that the results must be a conservative prediction, rather than a more realistic prediction, of the seismic load in the pipe.

What is the end product of this design-by-analysis approach for seismic support of piping systems? First of all, predicted seismic loads have increased by more than an order of magnitude since the early 1970s, and the design costs have increased in proportion. Piping systems are supported on the rigid side of the building response spectra to meet the piping stress criteria. This results in a large number of seismic supports on each system. Many of the supports must be snubbers to maintain flexibility for thermal effects. Because supports are designed for stiffness as well as strength, design for stiffness leads to formidable structures. Snubbers are a maintenance

problem. The large number of supports creates congestion, which impedes all maintenance activities.

There is another, hidden, problem with the design-by-analysis approach. The analytical predictions for these piping systems are very sensitive to changes in loading, changes in the physical configuration, or tolerances. A small change in a support location or a valve weight, a refined estimate of support stiffness, or another refinement in the analysis technique often led to unacceptable pipe stresses or support loads. This results in analysis and design iterations that are costly but do not add value.

The present approach to piping seismic design has been questioned within the industry since around 1980. Some of the concerns are expressed in NUREG-1061.¹² Reviews of the actual behavior of piping systems in earthquakes^{13,14} indicate that piping systems made of ductile material and not supported for seismic loads have survived significant seismic events. In addition, experimental dynamic tests of components and systems demonstrate that rupture or collapse is not a realistic failure mode for ductile piping and that piping is inherently rugged.

EXPERIMENTAL RESULTS

Many analytical and experimental investigations of seismic response of piping have been made over the last 15 years. Results from some of the experimental studies are summarized in Ref. 15. A brief overview of these experimental studies follows.

Japanese High-Level Tests

A series of high-level tests to determine the behavior of piping at high load levels was performed in Japan in the 1970s.¹⁶⁻¹⁸ Static tests of 8- by 6-in. and 8- by 4-in. branch connections indicated safety factors of from 4 to 10 in the Sec. III code prediction of primary stresses. Cyclic displacement tests of branch connections showed that extreme displacements are required to cause a fatigue failure in seven cycles. Vibration tests on piping spans with pressurized elbows at load levels that cause ratcheting indicated no significant effect of the ratcheting on fatigue life. Cyclic displacement tests on pressurized fittings for fatigue indicated cyclic lives greater than those predicted from the Sec. III Class 1 fatigue rules.

Westinghouse Hanford Tests

Dynamic tests on a 1-in., schedule 40, unpressurized, insulated stainless steel piping system representative of the Fast Flux Test Facility were reported.¹⁹⁻²¹ The first

test configuration was supported at 11 locations with 6 rigid struts and 14 mechanical snubbers. There was no visible damage at twice the SSE levels. The SSE response spectrum has a 4-g peak.

Supports were removed, and the loading was changed to a sinusoidal input at the first mode frequency. With an input load level of about 0.2 g (0.5-g pipe acceleration), a plastic hinge was apparently formed. Permanent local yielding and deformation were visually observed. With higher loadings the vertical portion of the pipe gradually rotated. Response measurements indicated a second plastic hinge at about 3-g pipe acceleration (1.5-g base input). Gross distortion was observed following the test.

The equivalent damping coefficient was near 10% when the structure was elastic and approached 50% as large deformations occurred (strain range of 1% or more). The piping was subjected to loads in excess of four times the level that would cause calculated stresses at the ASME Code Level D allowables (using 10% damping) before gross deformation occurred.

EPRI/NRC Piping and Fitting Dynamic Reliability Program

In 1985, the Electric Power Research Institute (EPRI) in conjunction with the NRC initiated a program to test pipe components and piping systems for high-amplitude dynamic loads.²²⁻²³ Thirty-seven components (elbows, tees, and reducers) and two piping systems were dynamically tested to failure. Seismic time history inputs, scaled up in amplitude, were applied. For the component tests, the high-level test response spectrum was around 20-g zpa (zero period acceleration). The components withstood these high input accelerations without collapse of the cross section. Cyclic peak-to-peak strains of 3.4% were developed. Repeated tests resulted in through-wall cracks from fatigue-ratcheting. Inelastic behavior resulted in equivalent damping of 34%. The failure modes were fatigue-ratcheting except for two of the component tests: an unpressurized 6-in. schedule 10 elbow and an unpressurized 8 by 4 reducer with attached 4-in. pipe. For these two test specimens, failure was attributed to incremental deformation with the inertia arm ending in a displaced position.

EPRI/NRC System Test 1

System 1 (Refs. 23 to 25) consisted of 6-in.-diameter schedule 40 piping with a 3-in.-diameter schedule 40 bypass line and an 18-in.-diameter schedule 30 vessel. The pipe was mounted on individual uniaxial shake tables, and the table input accelerations simulated earthquake

motion. Pipe material was ASTM A106B carbon steel. The piping system was pressurized to 1000 psi during the test. System 1 was designed for relatively balanced pipe stresses. Test levels are shown in Table 2. The designator of "5SSE" implies an input that is approximately five times as great as the input designated as SSE.

Nine tests were run at OBE, SSE, and 5SSE levels. At the 5SSE level, a small amount of diametric swelling was noted in the elbows near the shaker tables. During the half-table capacity test, the spring hanger failed, the connecting bolts on the motor-operated valve actuator broke, and the actuator fell off. There was measurable permanent deformation of the system and increased swelling at the elbows but no leakage.

During the full-table capacity test, a short radius elbow near one of the sleds failed at 10 seconds into the test. This elbow was subjected to essentially a torsion load. The failure appeared to initiate by fatigue-ratcheting at a location 90° from the crotch of the elbow in a manner

similar to the component test failures. The through-wall crack developed into a ductile tear before the test could be stopped. The failure was localized. The system did not leak or collapse before the failure, and flow through the system was not compromised before the failure. The piping successfully withstood the half-table loading of 16-g zpa, and limited ratcheting and limited permanent deformation occurred.

EPRI/NRC System Test 2

System 2 was a stainless steel (ASTM 316L with 316 mechanical properties) piping system.^{24,26} There were 52 ft of 6-in. schedule 40 pipe, and the ends rose from two sleds. System 1 was designed as a "balanced" system, and system 2 was designed as an "unbalanced" system. Testing was conducted with the pipe filled with oil at 1000 psi. The test series and results are summarized in Table 3.

At the SSE level input, a minor degree of ratcheting was measured on the 12-in. vessel. At the 2SSE level, the ratcheting and displacement amplitude increased by a factor of 4. The mid-frequency test caused significant strain at the 4 × 12 reducer weld at the vessel nozzle as a result of an 8-Hz peak in the mid-frequency response spectra. A 60-in./second velocity sine sweep caused significant ratcheting, and a surface crack was noted in the 4 × 12 reducer weld. The 5SSE test did not result in significant additional ratcheting. At one-half sled capacity, the snubber clevis bent, and the weld crack propagated. The snubber was replaced with a 6000-lb snubber. During the full-sled level test, failure occurred

Table 2 Test Levels for System Test 1

Test type ^a	Input acceleration
OBE	
SSE	0.4-g zpa and 4.5-g peak
5SSE	2.5-g zpa
Half-table capacity	16-g zpa
Full-table capacity	30-g zpa

^aOBE, operating basis earthquake; and SSE, safe shutdown earthquake.

Table 3 Test Data for System Test 2

Test type ^a	Input level	Cyclic strain (peak to peak), %	Ratchet strain, %	Equivalent, damping, ^b %
SSE	0.4-g zpa			
2SSE	0.8-g zpa	0.21	0.07	5
Mid-frequency	10 to 18 g			
Sine sweep	8 g (60 to 20 Hz)			
Sine sweep	50 to 60 in./second			
5SSE	2.5- to 3.0-g zpa	0.77	0.18	22
Half-sled	4.5-g zpa	0.96	0.65	35
Full-sled	7.0-g zpa	2.8	2.1	

^aSSE, safe shutdown earthquake.

^bDamping in response spectrum analysis that results in the same moment as the measured moment at the failure location.

when a crack propagated through the wall. Also, a bulge was observed in the 6-in. schedule 40 pipe above Sled 4. The mid-frequency test and the sine sweep test (60-in./second velocity) both caused significant ratcheting at the failure location.

System test 2 developed a through-wall crack during the full-sled output test at the 7-g zpa loading. However, the system had been subjected to high-level sine sweep tests at 8 g, which resulted in ratcheting and surface cracks before the full test was run.

ETEC 3-in. Piping Demonstration Test

A piping system consisting of 51 ft of 3-in. schedule 40 A106 carbon steel was tested²⁴ at the Energy Technology Engineering Center (ETEC) in the unpressurized condition in 1987. There were no supports on the system. The piping was subjected to three levels of seismic input:

1. Low level—5-g nominal zpa
2. Intermediate level—14-g nominal zpa
3. High level—30-g nominal zpa (35-g actual)

There were no structural failures. Three low-level harmonic constant displacement inputs were applied. Again there was no failure. Then a 6-Hz, ± 7.5 -in. sine burst test was run. Failure occurred at the tee during the sine burst test at about 25 g. The failure appeared to be a fatigue failure in the crotch area of the tee accompanied by local structural collapse. On the basis of a linear-elastic response spectrum analysis using 5% damping, the Class 1 Level D piping allowable load was 1.4-g zpa.²⁰

ETEC 6-in. Piping Feasibility Test

A 6-in. piping system with a 3-in. branch line was tested to failure in an NRC-sponsored fragility test^{24,27} at ETEC from 1987 to 1988. There was 48 ft of 6-in. piping and 17 ft of 3-in. piping. Pipe material was A106B carbon steel. The pipe was filled with oil and pressurized to 1000 psi for the test. The piping was subjected to the following three levels of seismic input (15-second duration).

1. Low level—5-g nominal zpa
2. Intermediate level—14-g nominal zpa
3. High level—25-g nominal zpa (30-g actual)

Failure did not occur during this seismic testing. However, plastic deformation was noted in the 10-ft vertical riser after the high-level test. A 1-in.-wide circumferential

bulge, indicative of ratcheting, was located about 2 1/2 in. above the welding neck flange "anchor." The measured diameter of the bulge was 6.83 in. in comparison with a measurement of 6.66 in. at 2 ft away.

Then two sine burst tests were run.

- Sine burst—4 Hz, 8 cycles of ± 7 -in. displacement, 12-g nominal, 18-g actual
- Sine burst—5 Hz, 11 cycles of ± 7 -in. displacement, 18-g nominal, 48-g actual

Rupture (a 300° circumferential break) occurred at the bulge during the sixth of the planned eleven cycles of the second sine burst test. Failure was attributed to incremental ratcheting, which resulted in wall thinning and subsequent fracture caused by tensile overloading. No evidence of fatigue failure was found on the metallurgical examination. On the basis of postfailure diametral growth and wall-thinning measurements, the average circumferential and radial residual strains in the failure zone were 9.2% and -12%. Longitudinal residual strain was 0.7%. Local wall thinning of up to 25% was found at one location along the fracture surface during posttest examination.

The allowable zpa seismic loading was 2 g on the basis of linear-elastic analyses with 5% damping, Class 1 Code analysis, and a $3S_m$ limit.²⁷ Tested seismic levels were 15 times higher (30 g/2 g) without failure.

BNL Vibration Tests

The Berkeley Nuclear Laboratories (BNL) in the United Kingdom has carried out a comprehensive test program on dynamic response of pipe.²⁸⁻³⁰ Lengths of pinned-end straight pipe were vibrated at or near their fundamental frequency. The main conclusion of this work was that yielding of the material limits the resonant response of the pipe when subjected to high levels of vibration. The response was self-limiting at a level that does not cause a low-cycle fatigue failure in straight pipe. Inelastic strain energy absorption significantly increased the apparent damping.

Vibration of pressurized specimens resulted in hoop ratcheting. The consequent accumulation of strain significantly hardened the material, increased the elastic strain range, and reduced the ratcheting. The measured elastic strain range more than doubled with the rapid accumulation of about 2% strain. The hoopwise ratcheting did not lead to failure (in these specimens). The rate of ratchet strain accumulation reduced significantly after the early cycles. Total accumulated strains of 5% after testing were typical.

Tests of pressurized straight pipe with a local thin section led to greater ratcheting and induced a ratchet failure that appeared to be caused by ductility exhaustion. A hoop stress equal to the yield stress, S_y , reduced the cyclic life in comparison to an unpressurized test by a factor of more than 10. A configuration with a large structural discontinuity (similar to back-to-back flanges) was also tested. The physical restraint of the discontinuity prevented significant hoop ratchet strain.

PIPING SEISMIC RESPONSE CHARACTERISTICS

The dynamic test data clearly demonstrate the inherent capability of ductile steel pipe to withstand strong earthquake motions. Piping systems and components subjected to extreme levels of simulated seismic motion survived without failure by collapse or fatigue. Elbows, tees, and reducers have been subjected to input motions with a response spectrum z_{pa} of up to 20 g. Experimental piping systems have been subjected to seismic motions with a z_{pa} of up to 35 g.

Collapse of thick-wall piping from seismic loading is not a realistic failure mode. This fact has been proved by the experimental studies of both components and piping systems with diameter-to-thickness ratios (D/t) of less than 50. Components were subjected to dynamic moments of twice the theoretical static collapse moment without failure. The reversing inertial load apparently does not act long enough in one direction to cause significant deformation or collapse. Thus the Code *primary stress limits* are excessively conservative for seismic loads.

The single most significant characteristic of the response of piping systems to dynamic input motions is inelastic energy absorption. Sinusoidal vibration tests of straight pipe segments demonstrate the self-limiting nature of the dynamic response. Yielding of the pipe has two effects. The natural frequency is shifted, and energy absorption attenuates the response. The effective damping increases greatly at high response levels. For piping component tests, effective damping has been calculated to be as high as 34%. For system tests, effective damping was from 13 to 23% at the highest levels. Effective damping for sinusoidal input motion with plastic hinges in the piping system was as much as 50%.

With constant internal pressure at a level that causes hoop stress at two-thirds of the yield stress, ratcheting occurs when the piping is subjected to cyclic bending moments above yield. The level of ratcheting in the

experiments is below 5%. At this level, there does not seem to be a measurable impact on the fatigue life.

CONCLUSIONS

The NRC regulations were needed by the industry in the early 1970s. These were times of rapid evolution of analysis methods and complexity of plant design. However, requiring linear-elastic dynamic analyses with low damping was inconsistent with the seismic technology at the time. The beneficial effects of plastic yielding for seismic response of structures had been recognized and studied since 1957. The NRC requirements were a conservative approach to seismic design. It could have been an acceptable approach if the final product—the piping system support design—had been both cost-effective and reliable. However, this was not to be!

A number of nuclear piping cost studies show a dramatic increase in the cost of the piping systems not only in absolute value but also relative to the total cost of building a nuclear power plant. For plants built during 1967 to 1974, the cost of safety-related piping, including materials, engineering, and construction, was about \$10M, or about 8% of the total plant cost. For plants built during 1981 to 1990, the cost of the safety-related piping had increased to about \$175M, or to about 15% of the total plant cost. Maintenance and replacement costs have also increased by about the same magnitude. Most of the cost increase is related to the additional engineering and design labor required to satisfy the seismic design criteria. Increased reliability, if any, has not been quantified.

One major problem is the emphasis on analysis rather than design. Each step in the analytical process and every aspect of the design procedure have been carefully scrutinized. Each step is evaluated on the basis of whether the analytical results are conservative and justifiable in comparison with linear-elastic time history analyses using the latest techniques. Resolution by linear-elastic analyses of concerns on design aspects, such as support gaps and bilinear snubber spring constants, has been attempted. The technical issue has become the adequacy of the linear-elastic analysis methods rather than the adequacy of the design to withstand the seismic event. Experimental results clearly demonstrate that piping does not respond in a linear-elastic manner during an earthquake. There are all kinds of nonlinearities. We keep on refining our linear-elastic analysis methods to get a "better" numerical result, but we are not doing a better job of representing what actually happens in an earthquake.

It is evident that our present approach to piping seismic response is not cost effective, practical, or reliable

and does not result in a better design. Looking back, it is apparent that the NRC requirements were a necessary step in an evolutionary process. The standardization that resulted was beneficial to the entire industry. However, the refinements—always striving to get more rigorous and conservative analytical predictions—are misguided from the standpoint of achieving the necessary level of safety at a reasonable cost. Value is not being added, and the costs are a burden.

Basing acceptability criteria on linear-elastic analysis methods is inconsistent with reality. The beneficial aspects of inelastic energy absorption in ductile steel piping are not used in the present methods of seismic analysis of piping systems. Linear-elastic response spectrum methods with low damping (5% or less) overpredict piping response if the pipe yields.

Sufficient data now exist to support a fundamental change in the way that piping systems are engineered to reliably withstand earthquakes in a cost-effective manner. Activities within the industry, to date, have been aimed at fixing small parts of the process. In the authors' opinion, real progress will not be made until an action plan is developed that includes all aspects of the problem, including cost and necessary level of protection. If such an action plan were available and agreed to by all concerned parties, the elements of the plan could be developed and implemented to yield the most value per unit cost. This planned approach would also allow time for both the industry and the NRC to adapt to new ways of doing business.

One step in the process of doing things differently is to revise the piping stress limits in the Sec. III code for seismic loadings. As of April 1994, the ASME Boiler and Pressure Vessel Code Committee has been considering a proposal to increase the Level D primary stress limits by 50%.

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Environmental Effects

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PC-Based Probabilistic Safety Assessment Study for a Geological Waste Repository Placed in a Bedded Salt Formation

By S. A. Khan^a

Abstract: A probabilistic safety assessment study is performed for a repository placed in a bedded salt formation using a fault tree analysis approach and the capabilities of IRRAS-PC code. The sensitive areas in repository safety are identified, and results for different risk importance measures are reported. Two release scenarios based upon human intrusion are also studied. The impact of both intact and degraded institutional control on risks to future generations is assessed. As a result of this analysis, the important events for future investigations are identified as faulting, groundwater movement, human activities, and erosion. The less important events are identified as diapirism, intrusive magmatic activity, second-order slumping, second-order denudation, second-order erosion, and third-order meteorite impact. It is concluded that under the normal geological evolution process, the failure probability of a repository in a salt formation is very low (2.63×10^{-5} in 1.0×10^5 years). However, if institutional control degrades, then chances of release of waste are higher. It is concluded that FTA presents the best approach for increasing the public confidence in long-term safety of geological waste repositories. Besides, the IRRAS-PC code is a powerful tool in predicting the long-term risk of high-level nuclear waste disposal in most of the candidate rock formations.

Isolation of high-level nuclear waste in deep geological rock formations through the use of a multibarrier approach is an attractive waste disposal concept (Fig. 1). However, the effectiveness of waste isolation in deep rock formations cannot be verified or disproved on the

basis of the experimentation or operating experience because of the very long time periods involved (e.g., million of years). It is generally believed that confidence in long-term safety of geological repositories can be based on the predictive modeling of a disposal system.¹⁻⁴

The probabilistic safety assessment (PSA) of radioactive waste disposal has gained a wide acceptance internationally over the past 5 to 7 years.⁵ In the context of repositories, PSA generally aims at predicting the doses and their probabilities to future generations at different time intervals. Some of the analyses performed so far are reported in Refs. 2 to 5.

When a PSA is performed for a repository, the fault tree analysis (FTA) technique is effective in predicting the release probability of waste to the biosphere (top event) in terms of the failure of barriers in a waste isolation system (components).³⁻⁹ The application of FTA, following a strict logical scheme, enables researchers to identify all the conceivable release scenarios (cut sets) and bring them together in a coherent system.

The work presented in this article describes the prediction of the overall failure probability of a waste disposal system in a salt formation. A time interval of 1.0×10^5 years is chosen because of more complete data availability for this time interval. Besides, this is a sufficiently long time to allow geological phenomena to develop to a significant level. The IRRAS-PC code¹⁰ is used for this purpose. The code utilizes the FTA approach and generates various reports as desired by the user. The code can assess both qualitatively and quantitatively the important

^aConsulting Engineer, Sak Engineering, Islamabad, Pakistan.

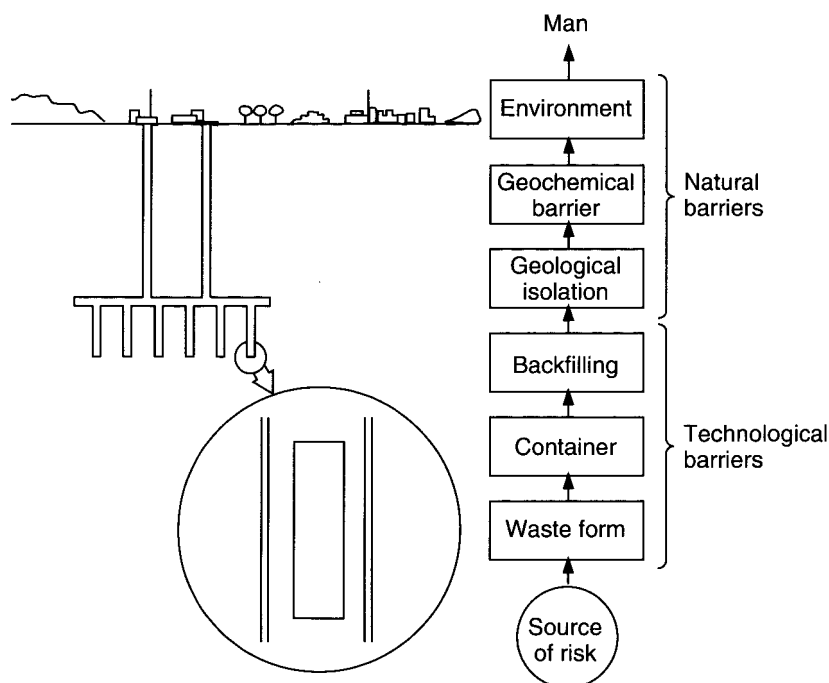


Fig. 1 Simple model of the geological repository concept.²

parameters related to the safety of a system under analysis. Details may be found in Ref. 10.

The work presented here has four objectives: (1) to test and validate the capabilities of IRRAS-PC code for the PSA of a repository by using an FTA approach, (2) to evaluate the various risk importance ratios for the disposal system, (3) to identify the important contributors to the release phenomena, and (4) to evaluate the impact of uncertainties in human intrusion (HI) values on waste release. For this purpose, three release scenarios are analyzed, one as a reference case⁴ and the other two as limiting cases.

For objectives 1 and 2, the FTA and data from Ref. 4 are used to analyze the reference case. Results are compared with those reported in Ref. 4. The comparison has enhanced the confidence in the prediction of the IRRAS-PC code. For objectives 3 and 4, two scenarios are analyzed. In scenario 1, the assumption is made that current institutional control is made more effective with the passage of time (limiting case 1). In scenario 2, the assumption is made that current institutional control degrades in an interval of 1.0×10^5 years, and almost no control remains effective (limiting case 2). The safety assessment of waste disposal is a multifaceted problem. It has a broad scope and involves a number of parameters in the repository failure process; therefore this study may be treated as a limited-scope study.

FAULT-TREE ANALYSIS DESCRIPTION

Fault-tree analysis is based on Ref. 4. The tree is developed for a bedded salt repository placed at a depth of 300 m. The FT consists of three basic branches, which show the geological containment failure mechanism:

1. Exhumation
2. Flooding by water
3. Meteorite impact

Details of each of the sections are shown in transfer gates (trees) in Figs. 2 to 5. In these figures, these three types of gates are used: OR gate (with "+" sign), AND gate (with "x" sign), and transfer gate (with "Δ" sign).

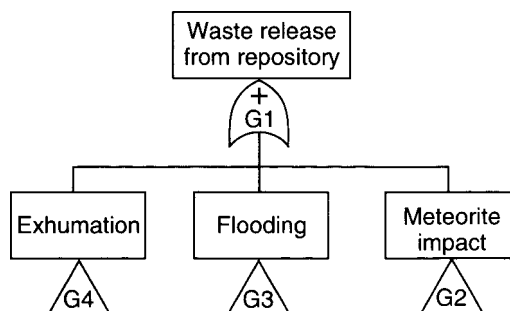


Fig. 2 Fault tree for salt repository.

data having inconsistent formats from different sources. Estimates of the future probability of a geological event that had previously occurred at a site only rarely or never in the history of the earth are highly uncertain. Therefore a more reliable estimation of the conditions of buried waste after thousands of years would require more extensive data than those now available.

The third area is the difficulty that stems from the concept of failure rate. Because the failure is not random in the geological evolution process, the concept of failure rate is not as useful (e.g., in the context of failure of engineered and natural barriers in a repository) as the component failure rate in other nuclear systems. Barriers in a geological disposal system can either act to prevent the initiation of radionuclide release or retard the rate at which the release occurs. Thus barriers in the system do not experience failure (in its common meaning), and the term can be used only in a sense that a barrier does not meet the standard set for its performance.

For a bedded salt formation,^{4,11-18} the potential mechanism for containment failure (i.e., the basic events) may be the anthropogenic causes (i.e., sabotage, nuclear warfare, or drilling) or natural causes (i.e., meteorite impact, volcanism, faulting, or erosion). Some studies¹¹⁻¹⁸ indicate that volcanism and meteorite impact may simply be neglected for a detailed analysis in the salt formation. Whereas a water intrusion scenario is considered the dominating cause of release, some experts¹⁸ believe that for salt formations the possibility of a serious breach of containment of waste repository by natural or human

events is extremely remote. It is believed that a sealed repository would be sabotage-proof, and even a 50-MT nuclear explosion would not breach the containment.

HUMAN INTRUSION

Human intrusion is one major area of uncertainty. Over very long time periods, HI at a waste disposal site is a major threat to the repository integrity as institutional control becomes less effective and future generations find warning signs difficult to decipher. An HI scenario is equal in potential to the groundwater (GW) release scenarios as far as the radiological consequences are concerned. The earlier HI analyses have been mostly performed for disposal facilities for low-level waste (LLW) and specially for salt formations.^{19,20}

Human intrusion at a repository site may be inadvertent human actions (e.g., various mining operations, large engineering operations, drilling for water or exploration, and underground explosions such as nuclear tests) in which the intruder is unaware of the presence of waste or its potential hazard. HI may also be due to deliberate actions, such as sabotage, terrorism, or war. Different scenarios may be imagined for the intrusion at a disposal site. Various approaches have been used for the quantification of HI probabilities (e.g., event trees,¹⁻² product integral approach,²¹ and data based on subjective-expert judgment).¹ Typical HI data for a salt and plastic clay formation are reported in Tables 1 and 2, respectively.

Table 1 Basic Event Probability Values for Human Actions for Salt Deposits—Direct Breach (Exhumation)^a

Factor	Time, years			
	1.0×10^3	1.0×10^4	1.0×10^5	1.0×10^6
Loss of memory	- (1.0×10^{-2})	1.0×10^{-2} (0.2) ^b	0.5 (0.5)	1.0 (1.0)
Interest in mining	- (0.5×10^{-3})	1.0×10^{-3} (0.5×10^{-2})	1.0×10^{-2} (0.5×10^{-2})	1.0×10^{-2} (0.5×10^{-2})
Geometrical factor ^c	- (2.0×10^{-2})	2.0×10^{-3} (2.0×10^{-3})	2.0×10^{-3} (2.0×10^{-3})	3.0×10^{-2} (2.0×10^{-3})
Total	- (1.0×10^{-8})	2.0×10^{-8} (2.0×10^{-6})	1.0×10^{-5} (5.0×10^{-6})	3.0×10^{-4} (1.0×10^{-5})

^aBased on Ref. 4.

^bValues in parentheses are for the exhumation of waste.

^cGeometrical factor is the probability that a hole will be drilled specially in that area.

Table 2 Probability Range of Human Action Event (Drilling) that May Expose Waste to Aquifers^a

Factor	Time, years			
	2 000	25 000	100 000	250 000
Loss of memory	0.1 to 0.5	0.8 to 1.0	1.0	1.0
Interest in drilling	0.1 to 0.2	0.1 to 0.1	0.1 to 0.1	0.1 to 0.1
Geometrical factors	4.0×10^{-3}	4.0×10^{-3}	4.0×10^{-3}	4.0×10^{-3}
Drilling that may expose waste to aquifers	4.0×10^{-6} to 4.0×10^{-4}	3.2×10^{-5} to 4.0×10^{-4}	4.0×10^{-5} to 4.0×10^{-4}	4.0×10^{-5} to 4.0×10^{-4}

^aSee Refs. 1 and 3.

Table 3 Primary Event Probabilities for Bedded Salt Formations^a

Factors	Time, years			
	1.0×10^3	1.0×10^4	1.0×10^5	1.0×10^6
Meteorite impact	(1.0×10^{-10}) (1.0×10^{-10})	(1.0×10^{-9}) (1.0×10^{-9})	(1.0×10^{-8}) (1.0×10^{-8})	(1.0×10^{-7}) (1.0×10^{-7})
Faulting-water intrusion	1.0×10^{-7} (1.0×10^{-4})	1.0×10^{-6} (1.0×10^{-3})	1.0×10^{-5} (1.0×10^{-2})	1.0×10^{-4} (1.0×10^{-1})
Volcanic explosion	(1.0×10^{-9})	(1.0×10^{-8})	(1.0×10^{-7})	(1.0×10^{-6})
Volcanic transport to surface	(1.0×10^{-8})	(1.0×10^{-7})	(1.0×10^{-6})	(1.0×10^{-5})

^aSee Ref. 1. Values in parentheses are based on fault-tree-analysis. Other values are based on expert opinion.

It is obvious from these tables that loss of memory is almost certain for time periods greater than 1.0×10^4 years. However, values reported in Table 2 are comparatively higher than those in Table 1 for loss of memory.

BASIC EVENT DATA USED IN THE FAULT TREE

Most of the data (integrated probability values for 1.0×10^5 years) used in this analysis are based on Ref. 4. The data are derived from expert opinion and historical records and are mostly time dependent (e.g., Table 3). Most of the data presented here may be uncertain (e.g., HI values), and a variation of some order of magnitude may not be uncommon for probability values, depending upon the source on which they are based. Therefore an uncertainty or sensitivity analysis over a range of these

values may be desired, and the impact over the result of the study may be mentioned. However, such a detailed analysis is presently outside the scope of this paper and may be covered in the future.

The values reported in Table 4 are for flat-bedded salt about 300 m thick located in an evaporitic basin of 5000 km². The top of the salt bed is some 350 m below the land surface. For the hydrogeological situation of the repository, a single aquifer system is assumed to exist above the impermeable upper layer. Below the impermeable layer, saltwater and confined fresh water are also envisaged.⁴

RESULTS AND DISCUSSION

The PSA results, for salt repository using the IRRAS-PC code, are reported in Tables 5 to 8 for a time period of

Table 4 Primary Event Probabilities for Bedded Salt for a Time Period of 1.0×10^5 Years^a

Event description	Event notation in fault tree	Failure probability
Small displacement fault	E1	4.0×10^{-5}
Undetected fault	E2	6.0×10^{-10}
Healed fault	E3	4.0×10^{-3}
Revival of healed fault	E4	4.0×10^{-4}
Large fault	E5	4.0×10^{-6}
Very large displacement fault	E6	4.0×10^{-7}
Diapirism	E7	5.0×10^{-7}
Groundwater above	E8	1.0
Groundwater below confined	E9	0.5
Replacement by fresh water above	E10	0.2
Replacement from below	E11	0.3
Replacement in fractured layers	E12	0.5
Diapirism (uplift)	E13	4.0×10^{-7}
Human actions (top layers failure)	E14	1.0×10^{-5}
Exhumation by human actions	E15	5.0×10^{-6}
Intrusive magmatic activity (IMA)	E16	4.0×10^{-9}
Exhumation by volcanism	E17	4.0×10^{-9}
Slumping-I	E18	1.0×10^{-8}
Slumping-II	E19	1.0×10^{-7}
Denudation-I	E20	1.0×10^{-8}
Denudation-II	E21	1.0×10^{-8}
Erosion-I	E22	1.0×10^{-6}
Erosion-II	E23	1.0×10^{-5}
Meteorite impact-I ^b	E24	4.0×10^{-8}
Meteorite impact-II ^c	E25	1.3×10^{-7}
Meteorite impact-II ^d	E26	1.3×10^{-7}
Meteorite impact-III ^e	E27	7.0×10^{-7}
Meteorite impact-III ^f	E28	7.0×10^{-7}
IMA (layers failure)	E29	4.0×10^{-9}
Soil retention	E30	0.1

^aSee Ref. 4.

^bFirst-order meteorite impact (MI) can pulverize the wastes.

^cSecond-order MI can fracture the unmoved repository.

^dSecond-order MI can pulverize the partially uplifted repository.

^eThird-order MI can fracture the impermeable layers over the repository.

^fThird-order MI can fracture the partially uplifted repository.

1.0×10^5 years. The results include, for example, dominant cut sets, important risk ratios, uncertainty analysis, and scenarios sensitivity analysis. The important risk ratios (or importance measures) include the Birnbaum Importance, F-V Importance, Risk Achievement Worth Ratio, and Risk Reduction Worth Ratio. For a better understanding of the results presented in this article, different risk ratios are defined in the following paragraphs:^{22,23}

1. F-V Importance:

This is the fractional contribution of the *i*th component (event) to the risk of a system and is expressed as (*I_i*):

$$I_i = (R_o - P_i)/R_o$$

where *R_o* is the present-nominal risk level of the system and *P_i* is the decreased risk level with the component "i" optimized or assumed to be perfectly reliable.

Table 5 Important Cut Set Analysis Results for a Bedded Salt Repository (Reference Case)

Min Cut Upper Bound for the Top Event $\rightarrow 2.63 \times 10^{-5}$					
Accumulative probability, ^a %	Cut set contribution, ^b %	Failure probability	Cut sets (events) ^c		
30.37	30.37	8.00×10^{-6}	E1	E10	E8
53.15	22.78	6.00×10^{-6}	E1	E11	E9
72.13	18.98	5.00×10^{-6}	E15		
79.72	7.59	2.00×10^{-6}	E10	E14	E8
87.31	7.59	2.00×10^{-6}	E12	E5	E8
91.11	3.80	1.00×10^{-6}	E22		
94.90	3.80	1.00×10^{-6}	E12	E5	E9
96.42	1.52	4.00×10^{-7}	E6		
97.64	1.21	3.20×10^{-7}	E10	E3	E4 E8
98.55	0.91	2.40×10^{-7}	E11	E3	E4 E9
99.08	0.53	1.40×10^{-7}	E10	E27	E8
99.38	0.30	8.00×10^{-8}	E10	E13	E8
99.63	0.25	6.50×10^{-8}	E12	E25	E8
99.78	0.15	4.00×10^{-8}	E24		
99.90	0.12	3.25×10^{-8}	E12	E25	E9
99.94	0.04	1.00×10^{-8}	E18		
99.98	0.04	1.00×10^{-8}	E20		
99.99	0.02	4.00×10^{-9}	E17		
100.0	0.00	8.00×10^{-10}	E10	E29	E8

^aAccumulative cut set contribution to the repository system failure.

^bIndividual cut set percent contribution to the top event failure.

^cIn column 4, E1, E10, etc., mean event No. 1, No. 10, etc.

2. Risk Reduction Worth Ratio:

This is defined as the decrease in risk if the feature were assumed to be optimized or were assumed to be made perfectly reliable. For a component "i," it is expressed as (Di):

$$D_i = R_o/P_i$$

where R_o is the present-nominal risk level and P_i is the decreased risk level with the component "i" optimized or assumed to be perfectly reliable.

3. Risk Achievement Worth Ratio:

This is the ratio of the risk that results from the ith component failed to the nominal risk.

$$Q(i) = S_i/R_o$$

where S_i is the increased risk level without component "i" or with component "i" assumed failed.

4. Birnbaum Importance:

If the risk measure is defined to be the system unavailability or unreliability, then the Birnbaum Importance can be defined as

$$B_i = A_i - C_i$$

where B_i is Birnbaum Importance, A_i is the system unavailability with component "i" assumed failed, and C_i is the system unavailability with component "i" assumed working.

Birnbaum Importance is actually the probability of change in risk for a change in failure of the ith component or system of concern. This ratio identifies systems important to safety; however, it does not consider the likelihood of these systems failing. The risk achievement worth and risk reduction together are more informative than the Birnbaum Importance.

Table 5 shows that the first seven cut sets account for about 95% of the top event probability. The first cut set

**Table 6 Important Risk Ratios for the Bedded Salt Repository
(Reference Case)^a**

Event ^a	Failure probability ^b	Risk ratios			
		Fussell – Vesely	Reduction	Achievement	Birnbaum
E1	4.00×10^{-5}	5.31×10^{-1}	2.13	1.21×10^4	3.20×10^{-1}
E8	1.00×10^0	4.79×10^{-1}	1.92	1.00	1.26×10^{-5}
E10	2.00×10^{-1}	4.00×10^{-1}	1.67	2.60	5.27×10^{-5}
E9	5.00×10^{-1}	2.76×10^{-1}	1.38	1.28	1.45×10^{-5}
E11	3.00×10^{-1}	2.37×10^{-1}	1.31	1.55	2.08×10^{-5}
E15	5.00×10^{-6}	1.90×10^{-1}	1.23	3.80×10^4	1.00
E12	5.00×10^{-1}	1.18×10^{-1}	1.13	1.12	6.19×10^{-6}
E5	4.00×10^{-6}	1.14×10^{-1}	1.13	2.37×10^4	6.25×10^{-1}
E14	1.00×10^{-5}	7.59×10^{-2}	1.08	7.59×10^3	2.00×10^{-1}
E22	1.00×10^{-6}	3.80×10^{-2}	1.04	3.80×10^4	1.00
E4	4.00×10^{-4}	2.13×10^{-2}	1.02	5.41×10^1	1.40×10^{-3}
E3	4.00×10^{-3}	2.13×10^{-2}	1.02	6.29	1.40×10^{-4}
E6	4.00×10^{-7}	1.52×10^{-2}	1.02	3.80×10^4	1.00
E27	7.00×10^{-7}	5.31×10^{-3}	1.01	7.59×10^3	2.00×10^{-1}
E25	1.30×10^{-7}	3.70×10^{-3}	1.00	2.37×10^4	6.25×10^{-1}
E13	4.00×10^{-7}	3.04×10^{-3}	1.00	7.59×10^3	2.00×10^{-1}
E24	4.00×10^{-8}	1.52×10^{-3}	1.00	3.80×10^4	1.00
E18	1.00×10^{-8}	3.80×10^{-4}	1.00	3.80×10^4	1.00
E20	1.00×10^{-8}	3.80×10^{-4}	1.00	3.80×10^4	1.00
E17	4.00×10^{-9}	1.52×10^{-4}	1.00	3.80×10^4	1.00
E29	4.00×10^{-9}	5.31×10^{-5}	1.00	1.21×10^4	3.20×10^{-1}
E2	6.00×10^{-10}	7.97×10^{-6}	1.00	1.21×10^4	3.20×10^{-1}
E23	1.00×10^{-5}	1.96×10^{-6}	1.00	1.20	5.16×10^{-6}
E28	7.00×10^{-7}	2.12×10^{-7}	1.00	1.30	7.96×10^{-6}
E7	5.00×10^{-7}	2.04×10^{-7}	1.00	1.41	1.08×10^{-5}
E26	1.30×10^{-7}	7.21×10^{-8}	1.00	1.55	1.46×10^{-5}
E19	1.00×10^{-7}	1.96×10^{-8}	1.00	1.20	5.15×10^{-6}
E21	1.00×10^{-8}	1.96×10^{-9}	1.00	1.20	5.15×10^{-6}
E16	4.00×10^{-9}	1.57×10^{-9}	1.00	1.39	1.03×10^{-5}

^aEvent in the first column indicates the primary event in the fault tree.

^bFailure probability for the primary events in 1.0×10^{-5} years.

(E1-E10-E8), which consists of the faulting phenomena in the presence of GW above or below the salt formation and which after saturation is continuously replaced by fresh water and is continuously dissolving the salt, weighs more than 30%.

The second cut set (E1-E11-E9), which consists of faulting phenomena coupled with confined groundwater below the formation and simultaneous replacement of this water, weighs about 23%. Both of these cut sets

indicate the critical potential of faulting phenomena coupled with the GW attack and its subsequent transport as a waste dispersion medium.

The results for the cut set obtained are in agreement with those reported in Ref. 4. In context with the fault tree analyzed in this paper, the agreement with the results reported in Ref. 4 enhances the confidence in code predictions. However, all output parameters from the code cannot be compared with those of Ref. 4.

**Table 7 Uncertainty Analysis Results for the Salt Repository
(Reference Case)**

A Monte Carlo Procedure for Determining the Distribution and Simulation Limits				
Random seed	3571			
Sample size	1000			
Number of events	61			
Number of cut sets	43			
Point estimate value	2.6343×10^{-5}			
5th Percentile value	1.9402×10^{-6}			
Median value	8.9132×10^{-6}			
Mean value	2.3510×10^{-5}			
95th Percentile value	8.7786×10^{-5}			
Minimum sample value	4.9075×10^{-7}			
Maximum sample value	1.1672×10^{-3}			
Standard deviation	5.6663×10^{-5}			
Coefficient of skewness	1.0795×10^1			
Coefficient of kurtosis	1.8240×10^2			

Distribution quantile level, %	95% Confidence interval on quantile level, % (+/-)	Quantile value	95% Confidence interval on quantile	
			Lower	Upper
0.5	0.5	6.3377×10^{-7}	4.9075×10^{-7}	8.3425×10^{-7}
1.0	0.7	9.9594×10^{-7}	5.7541×10^{-7}	1.1798×10^{-6}
2.5	1.0	1.4376×10^{-6}	1.1262×10^{-6}	1.7628×10^{-6}
5.0	1.4	1.9402×10^{-6}	1.8237×10^{-6}	2.0965×10^{-6}
10.0	1.9	2.4703×10^{-6}	2.2428×10^{-6}	2.7012×10^{-6}
20.0	2.5	3.6944×10^{-6}	3.3782×10^{-6}	3.9455×10^{-6}
25.0	2.7	4.2453×10^{-6}	3.9401×10^{-6}	4.7179×10^{-6}
30.0	2.9	5.1015×10^{-6}	4.5731×10^{-6}	5.7058×10^{-6}
40.0	3.1	6.9396×10^{-6}	6.4468×10^{-6}	7.4361×10^{-6}
50.0	3.1	8.9132×10^{-6}	8.0377×10^{-6}	9.7480×10^{-6}
60.0	3.1	1.2109×10^{-5}	1.1012×10^{-5}	1.3327×10^{-5}
70.0	2.9	1.6737×10^{-5}	1.5410×10^{-5}	1.8623×10^{-5}
75.0	2.7	2.0396×10^{-5}	1.8384×10^{-5}	2.2847×10^{-5}
80.0	2.5	2.6487×10^{-5}	2.2755×10^{-5}	3.1463×10^{-5}
90.0	1.9	5.1853×10^{-5}	4.3280×10^{-5}	6.0811×10^{-5}
95.0	1.4	8.7786×10^{-5}	7.7374×10^{-5}	1.0854×10^{-4}
97.5	1.0	1.3289×10^{-4}	1.0888×10^{-4}	1.7934×10^{-4}
99.0	0.7	2.0370×10^{-4}	1.7934×10^{-4}	4.1974×10^{-4}
99.5	0.5	3.4429×10^{-4}	2.3203×10^{-4}	1.1672×10^{-3}

The observations from Table 5 regarding the cut sets' generation and their weight are in agreement with the common opinion that GW attack and transport are the critical events in repository safety. In addition to the cut

sets Nos. 1 and 2, other ways of release are described by cut set No. 3 (exhumation by human action), No. 4 (removal of top layer by human action coupled with GW attack from above and simultaneous replacement of brine

**Table 8 Results for the Releases from a Bedded Salt Repository
(Reference Case and Scenarios 1 and 2)^a**

Results	Reference case	Scenario 1 ^b	Scenario 2 ^c
Release probability from the salt formation	2.63×10^{-5}	1.94×10^{-5}	5.22×10^{-3}
Most important event identified	E1 ^d	E1	E15
Fussell-Vesely of event	5.31×10^{-1}	7.23×10^{-1}	9.58×10^{-1}
Risk reduction	2.13	3.61	2.38
Risk achievement	1.21×10^4	1.65×10^4	1.92×10^2
Birnbaum	3.20×10^{-1}	3.20×10^{-1}	1.10
First dominating cut set	E1-E10-E8	E1-E10-E8	E15
Percent contribution	30.37	41.29	95.82
Second dominating cut set	E1-E11-E9	E1-E11-E9	E10-E14-E8
Percent contribution	22.78	30.97	3.83
Third dominating cut set	E15	E12-E5-E8	E1-E10-E8
Percent contribution	18.98	10.32	0.15
Human intrusion			
E14 values	1.0×10^{-5}	1.0×10^{-7}	1.0×10^{-3}
E15 values	5.0×10^{-6}	1.0×10^{-8}	5.0×10^{-3}

^aThe parameters at serial numbers 2 to 6 indicate the maximum values.

^bIn limiting scenario 1, the human intrusion chances are taken as very remote because of intact effective institutional control.

^cIn limiting scenario 2, effective institutional controls are assumed to be lost, and this value is taken as 2 orders of magnitude greater than those reported in the literature.

^dFor the description of the events E1, E2, etc., see Table 4.

with fresh water), and No. 5 (a large fault coupled with GW attack from above and its replacement with fresh water in the fractured layers).

Direct exhumation by human action [cut set No. 3 (E15), weight of 19%] and by erosional process [cut set No. 6 (E22), weight of 3.8%] are alternate modes of failure of some concern. The contribution of various cut sets to the overall probability of release is, however, quite different, as can be seen from Table 5. It is apparent that the probability of a release occurrence within 1.0×10^{-5} years after repository closure is related to the human activities and consists of direct exhumation of waste. This means that the most probable way in which waste would be released is the possibility that future generations, having forgotten the existence and potential danger of the repository, will start mining the salt formation.

Table 6 indicates the risk importance measures for various events. The first three events have the major risk reduction potential, that is, small displacement fault (E1), GW above the salt formation (E8), and replacement of saturated GW with fresh water (E10). These events have a risk reduction potential of 2.1, 1.9, and 1.7, respectively. This implies that, if the repository site is selected

so that future improvements on various activities reduce the possibility of developing small faults, GW above or below salt and exchange with fresh water lying above or below it, then risk of waste release to the future generation can be reduced by a factor of 2.1, 1.9, and 1.7, respectively, by such improvement activities in the time period of 1.0×10^{-5} years.

The risk achievement worth ratios indicate that, if sufficient efforts are not made to control the degradation of the repository in a salt formation because of such events as a very large displacement fault (E6), exhumation by human actions (E15), or erosion (E22), then the risk to future generations may increase by a factor of 3.84×10^{-4} at maximum. This fact underlines the importance of taking into account these factors in a repository site-selection process.

As shown in Table 6, future risk analysis activities can easily be prioritized in light of this analysis, and the important or unimportant events can be underlined for future research. For decision makers, the risk reduction and risk achievement potential of various events may be an important index of prioritizing the events in the repository site-selection procedure. For example, Table 6 shows

that when we ignore or do not take into account events like diapirism, above and below confined GW, and its replacement from above and below (i.e., events E7, E8, E9, E10, and E11), the present level of risk to the future generation is not significantly affected. Therefore the need for more precise evaluation of such events is, in fact, obviated. The events of less importance for future research seem to be E7, E16, E19, E21, E23, E26, E28, etc.

Uncertainty analysis results are reported in Table 7 for the maximum, minimum, point estimate, fifth percentile value, median value, and mean value. A Monte Carlo procedure is used to calculate the simulation limits.¹⁰ Some of the event values, such as HI, may have an uncertainty of some order of magnitude in their values. Therefore sensitivity or uncertainty analysis may be desired over a range of the values of these uncertain parameters. This is not currently in the scope of this analysis, but it may be covered in the future.

Results for the reference case (see Ref. 4) scenarios 1 and 2 are reported in Table 8. As already mentioned, the reference case defines the failure of the bedded salt repository as modeled by the fault tree reported in Ref. 4, and the actual HI data are based on the literature. Scenario 1 defines the failure of the repository with the completely degraded institutional control over the 1.0×10^5 years. Scenario 2 defines the same but with intact institutional control.

Table 8 shows that improving the possibility of *nil human intrusion* (e.g., strengthening the institutional control) at a repository site would not significantly reduce the risk of failure in 1.0×10^5 years. However, increasing the probability of HI at a repository site because of degradation of current practices would increase the risk of repository failure by a factor of about 1.8×10^2 . Those practices include institutional control; isolated or deserted sites selection; site reservation for public parks; archive records; and redundancy in keeping records, which may reduce the E14 (i.e., human action causing top layers failure) by a factor of 2 to 3 orders of magnitude. This increased risk underlines the importance of properly maintaining the current control practices if no further improvement is possible. It also indicates that improvement of the current practices has less potential for reducing risk; rather, it has high-risk achievement worths.

Tables 5 and 6 indicate that major scenarios of interest for a salt repository failure may be water flow at the edges, salt dissolution through percolating water caused by fracture of overlying protecting layers, and increase of water flow in the repository as a result of human activities. It is implied that, if a salt repository is placed in a volcanically, seismically, and tectonically stable zone,

other scenarios, such as meteorite impact and volcanism, may not need detailed analysis.

The important point emerging from this analysis is that, in terms of increasing the public confidence in realistic risks of nuclear energy, probabilistic safety assessment studies give useful results about the safety of waste disposal. Important information about prioritizing the areas for detailed analysis, screening the important and unimportant events, and establishing precise data requirements may be obtained in such an analysis.

It is also emerging that sensitivity and uncertainty analyses may be the best approaches for focusing on the really important and sensitive issues in the repository safety studies and increasing the public confidence in results. The sensitivity analysis may be for different scenarios (as performed in this article), or it may be over a range of values of different parameters. As a result, more sensitive areas may be defined for precise evaluation of the parameters. Besides the priority areas for detailed consideration in site selection, evaluation and other factors can be established from such an analysis on a case-by-case basis.

Much work needs to be performed on the different aspects of repository safety analyses, such as data, scenarios, and uncertainty and sensitivity analyses. This article is an important step forward in that direction and covers a limited-scope study for the sensitivity analysis of HI scenarios only. Before important decisions are made, however, this analysis indicates that a fault-tree model should be adequately described, the event dependencies should be adequately understood, and the processes involved in the repository failure should be well defined to eliminate the chances of any modeling errors.

It is a matter of fact that data uncertainties (of some order of magnitude) may be expected in some of the events and need to be resolved to the maximum possible extent. However, this study has demonstrated that the IRRAS-PC code is a useful tool to quantify the impact of such uncertainties on the failure of a repository system. Future activities, therefore, may be focused for a detailed analysis in that direction.

CONCLUSIONS

On the basis of the analysis, the following conclusions are drawn:

1. The IRRAS-PC code is a powerful tool for analyzing important issues on the prediction of long-term safety of geological waste repositories placed in salt and other rock formations.

2. Large faults, HI, and erosion are important single contributors to repository failure in a bedded salt formation. Over a period of 1.0×10^5 years, these events contribute more than 23% to repository failure.

3. Events that may combine to cause repository failure (cut sets) are GW movement coupled with small faulting. They contribute more than 50% to the repository failure.

4. The maximum risk reduction potential is for small faulting events. If the areas having faulting potential are avoided, risk to future generations may be reduced by a factor of greater than 2.

5. The maximum risk achievement potential is for such events as HI, erosion, and very large faults. These events have a potential of 3.8×10^4 .

6. The sensitivity analysis results indicate that, if the institutional control is intact for a very long time, then the most important event to the repository failure is small faulting. Under such a situation, the dominating cut set is a combination of the faulting with water movement, and the cut set contributes 41.3% to the repository failure. If the institutional control is lost, however, then the dominating cut set is human exhumation. This event alone can cause 95.8% of the repository failure.

7. Some events are identified as unimportant to the repository system failure in the long run; therefore they can be ignored in future repository studies for bedded salt formation (e.g., meteorite impact of the second and third orders, second-order erosion, diapirism, second-order denudation, and intrusive magmatic activity).

8. Important events are small faulting, GW movement, HI, and erosion.

9. There are some real limitations on collecting data for geological events and HI. Therefore, results need to be understood and evaluated in this perspective.

10. The safety analysis presented here implies that, because of uncertainties in the available data, sensitivity and uncertainty analyses may be the best approaches for increasing the public confidence in the risk assessment studies.

11. The fault-tree approach seems to be the best predictive method for the PSA of the geological waste repositories. However, the quality of results seems to depend largely on the quality of the input data for the basic events.

RECOMMENDATIONS

1. Comparative risk evaluation in different types of host rocks (salt, granite, clay, and tuff, for instance) is recommended for better risk understanding and decision making regarding the repository siting.

2. A priority area in future risk evaluation is a better understanding of the faulting, GW, and HI. For this purpose, comparison exercises of codes may be initiated.

3. Better communication of the risk to the public in light of the uncertainties in current models and data is needed.

4. Detailed uncertainty and sensitivity analyses are recommended for important events in the repository failure mechanism.

5. Highly active nuclear waste disposal is a global problem. A better exchange of knowledge, experiences, and ideas on the international level would be more fruitful.

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Operating Experiences

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Managing Aging in Nuclear Power Plants: Insights from NRC's Maintenance Team Inspection Reports^a

A. Fresco and M. Subudhi^b

Abstract: Age-related degradation is managed through the maintenance program of a nuclear plant. From 1988 to 1991, the Nuclear Regulatory Commission (NRC) evaluated the maintenance program of every nuclear power plant in the United States. The authors reviewed 44 out of a total of 67 of the reports issued by the NRC on these in-depth team inspections. The reports were reviewed for insights into the strengths and weaknesses of the programs as related to the need to understand and manage the effects of aging on nuclear plant structures, systems, and components. The authors' conclusions follow:

- Differing maintenance philosophies, financial resources, and the lack of regulatory requirements had an impact on plant management's attention to aging concerns.
- Separate programs that specifically address the management of aging were not noted.
- Weaknesses existed in some portions of maintenance programs deemed important for understanding and managing aging, whereas other programs were strong or in the process of being strengthened.
- Maintenance programs rated "good" or "satisfactory" did not necessarily address adequately concerns related to aging-related degradation.
- Improvements in preventive and predictive maintenance programs, including failure trending, root cause analysis, and an integrated maintenance data base, can significantly improve the management of aging degradation and the safety of nuclear plant operations.

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^bBrookhaven National Laboratory, Department of Advanced Technology, Engineering Technology Division, Upton, New York 11973.

Assuring the safe operation of a nuclear power plant depends, to a large extent, on how effectively one understands and manages the age-related degradation that occurs in structures, systems, and components (SSCs). During the plant's original licensing process, the utilities and the Nuclear Regulatory Commission (NRC) use all available sources, including equipment qualification (EQ) results, industry standards and practices, and vendor recommendations, to ensure that all SSCs remain able to accomplish their design functions during the life of a plant. Industry standards specify the requirements for utility EQ programs for selected safety-related SSCs, and they provide that the qualified life of a component can be based on the periodic surveillance-maintenance, test, and replacement-refurbishment recommendations based on documented data combined with the equipment service conditions and application criteria. These practices include periodic testing and inspection, replacement and refurbishment, condition monitoring, trending, reconditioning and lubricating, and performing advanced testing for early detection of incipient failures.

NRC Regulatory Guide 1.89, "Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants," adds that

Periodic surveillance and testing programs are acceptable to account for uncertainties regarding age-related degradation that could affect the functional capability of equipment. Results of such programs will be acceptable as ongoing qualification to modify designated life (or qualified life) of equipment and should be incorporated into the maintenance and refurbishment/replacement schedules.

After more than two decades of experience, the commercial nuclear power industry has many sources of information, such as regular NRC inspections, 10 *CFR* Part 21 reports by vendors, NRC Generic Letters, Bulletins, Information Notices, and research activities, including the Nuclear Plant Aging Research (NPAR) Program and the Nuclear Plant Reliability Data System administered by the Institute for Nuclear Power Operations (INPO). These sources have confirmed that failures of SSCs, even safety-related items, do occur. In recognition of this fact, the NRC implemented a team inspection program to evaluate and assess the current maintenance practices in place at all nuclear power-plant facilities.

From 1988 to 1991, the NRC staff conducted Maintenance Team Inspections (MTIs) at commercial nuclear power plants to evaluate the effectiveness of licensee maintenance activities and to determine the need for a maintenance rule.

In the current study, the reports issued by the NRC, which documented the results of the inspections performed by the NRC, were valuable resources of new information that could contribute significantly to the knowledge base of the NPAR Program. The NRC inspections were performance based and directed toward evaluating equipment conditions; observing in-process maintenance activities; reviewing equipment histories and records; and evaluating performance indicators, maintenance control procedures, and the overall maintenance program. The NRC teams selected certain systems and directed the inspection toward determining whether those systems were being properly maintained. In addition, the teams assessed whether the current maintenance activities would ensure proper function in the remaining life of the plant.

There are a total of 67 MTI reports, one for each site. For the purpose of this research, a representative sample of 44 reports, which were issued through the end of 1990 and were readily available for our study, was selected. These 44 reports correspond to 29 Westinghouse pressurized-water-reactor (PWR) units, 16 Combustion Engineering PWR units, 1 Babcock & Wilcox PWR unit, and 22 General Electric boiling-water-reactor (BWR) units. The reports themselves are comprehensive documents, some of which may be 70 or more pages long. The inspections were conducted with the use of the guidance provided in NRC Temporary Instruction 2515/97, "Maintenance Inspection Guide," dated Nov. 3, 1988, which includes a Maintenance Inspection Tree. The tree is based on the management oversight and risk tree (MORT) analysis methodology. Most, if not all, of the inspections were performed by different teams of NRC

inspectors so that the same team usually did not perform more than one inspection. The selection of systems inspected was also different from one report to another.

On July 10, 1991, the NRC did, in fact, publish 10 *CFR* 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." The rule is to become effective on July 10, 1996. The commission noted in the *Federal Register* that there is a clear link between effective maintenance and safety as it relates to such factors as number of transients and challenges to safety systems and the associated need for operability, availability, and reliability of safety equipment. Good maintenance also provides assurance that failures of other than safety-related SSCs that could initiate or adversely affect a transient or accident are minimized and that such an approach is consistent with the defense-in-depth philosophy. Maintenance is also important to ensure that design assumptions and margins in the original design basis are either maintained or not unacceptably degraded.

The Commission further noted that the results of the MTIs indicated that licensees have adequate maintenance programs and have exhibited an improving trend in program implementation. However, some common maintenance-related weaknesses were identified, such as inadequate root-cause analysis (RCA), which led to repetitive failures; lack of equipment performance trending; and the consideration of plant risk in the prioritization, planning, and scheduling of maintenance. In general, as evidenced by plant operational performance data and the results of NRC assessments, the industry has exhibited a favorable trend in maintenance performance. Nevertheless, the necessity for ongoing results-oriented assessments of maintenance effectiveness is indicated by the fact that, despite significant industry accomplishment in the areas of maintenance program content and implementation, plant events caused by the degradation or failure of plant equipment continue to occur as a result of ineffective maintenance. Additionally, operational events have been exacerbated by or resulted from plant equipment being unavailable because of maintenance activities.

In its summary in the *Federal Register*, the Commission stated its belief that to maintain safety it is necessary to monitor the effectiveness of maintenance and take timely and corrective action, where necessary, to ensure continuing effectiveness of maintenance for the lifetime of nuclear power plants, particularly as plants age. The rule requires that licensees monitor the performance or condition of certain SSCs against licensee-established goals in a manner sufficient to provide reasonable assurance that those SSCs will be capable of performing their intended functions. Such monitoring may take into

account industry-wide operating experience. Where monitoring proves unnecessary, the licensees are permitted to rely upon an appropriate preventive maintenance program. The licensees are required to evaluate the overall effectiveness of their maintenance programs on at least an annual basis, again taking into account industry-wide operating experience, and adjust their programs where necessary to ensure that the prevention of failures is appropriately balanced against the unavailability of the SSCs. For monitoring and maintenance activities that require taking equipment out of service, licensees should assess the total plant equipment that is out of service and determine the overall effect on the performance of safety functions.

Although the maintenance rule does not take effect until July 10, 1996, the NRC has issued *Regulatory Guide 1.160*, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." The guide states that NUMARC 93-01 provides methods acceptable to the NRC staff for complying with the provisions of 10 *CFR* 50.65.

METHODOLOGY

The major areas of utility maintenance programs that were evaluated by the NRC included (1) overall plant performance related to maintenance, (2) management support of maintenance, and (3) maintenance implementation. For the current study, we compiled and sorted diverse information reflecting those elements of a good maintenance program which can also effectively manage aging in nuclear power plants:

- Specific aging-related insights or management responsiveness to aging concerns.
- Preventive maintenance and incorporation of manufacturers' recommendations.
- Predictive maintenance and condition monitoring techniques.
 - Postmaintenance testing.
 - Failure trending analysis.
 - RCA or failure analysis.
- Use of probabilistic risk assessment (PRA) in the maintenance programs.

Findings in these seven broad categories were based on the evaluation of the entire MTI report in light of (1) positive aspects or attributes; (2) observation of neutral aspects; (3) negative aspects or deficiencies; (4) failures, usually direct references to a specific system or component; and (5) violations identified by the NRC staff.

Our study was limited to an evaluation of the MTI reports issued by the NRC as a result of its site inspection of nuclear power-plant facilities. No attempt was made to discuss any of the findings either with the NRC inspectors or the utility personnel. Neither utility rebuttals to the original MTI reports nor NRC reinspection at certain plants were considered. Because of the nature of these MTI reports, the process of selecting systems at a particular site, and the inspection process itself, the following conditions should be considered while interpreting the results presented in this article:

- As a result of previously known problems and problems identified during the inspection, the NRC inspection teams placed different emphasis on some topics at one plant as compared with another plant. As a result of the specific inspection requirements of each plant, the MTI reports vary significantly in emphasis and detail placed on particular topics.

- A typical MTI report describes both positive and negative aspects of a utility's maintenance program. However, in the majority of cases, the negative aspects are described in greater detail. Positive aspects are often described in general terms and may be broad statements on a major topic.

- It was sometimes difficult to differentiate between positive aspects and observations or between deficiencies and observations. Sometimes an NRC inspector merely described the aspects of a program without indicating whether they were considered positive or negative.

- One MTI report was generated for each site. Some sites are multiunit, and the reactor types may also be completely different from one unit to the next. For data analysis purposes, because we often could not determine specifically which unit the NRC inspectors were referring to, the multiunit sites with different reactor types were counted under each reactor type. Although we did not anticipate any differences in maintenance effectiveness on the basis of reactor type, for the sake of completeness, the data were analyzed separately for each of the four reactor types (i.e., Westinghouse, Combustion Engineering, and Babcock and Wilcox PWRs and General Electric BWRs) and also jointly for all four reactor types taken as a unit.

With proper consideration of these conditions, the quantitative data provide very limited insights into the effects of maintenance on age-related degradation. The data did not show a clear relationship to the age of the plants, and no firm conclusions can be drawn from the data set. However, the large data base of textual information was extracted and evaluated to present it in a

perspective useful to those concerned with the management of age-related degradation of SSCs in nuclear power plants. The information lends itself more to qualitative rather than quantitative evaluation; therefore the focus of this article is on providing qualitative assessments of the programmatic areas and on discussing this same information from a system and component perspective.

On the basis of the results from this evaluation, we attempted to define effective aging management practices. The research recommendations from the NPAR studies on various SSCs provide a basic technical foundation in understanding, detecting, and mitigating age-related problems. Because a plant's maintenance program is the principal vehicle through which age-related degradation is managed, this article describes some of the organization and management factors that should be considered to implement each of the activities required in managing the effects of aging. Because all plants have infrastructures in place that can deal with the effects of aging, these organization and management factors can heighten utility awareness of the importance of age-related degradation and of the use of existing organizational assets to effectively detect and mitigate their effects.

PROGRAMMATIC INSIGHTS

These are programmatic aspects of effective aging management practices:

1. A clear understanding and recognition of aging of SSCs.
2. Identification of effective aging management practices that should be able to detect and mitigate the effects of aging at an incipient stage.
3. Management and organization attentiveness to aging.

The following discussions provide some of the characteristics of the status of the programmatic efforts by the utilities.

Specific Aging Insights

In general, the MTI reports provide substantial information on how plant maintenance programs address the aging of SSCs. This includes the attitude of management toward the aging issues and specific program attributes that address the detection or mitigation of degradation caused by aging.

Although some utilities appeared to assume a proactive stance to prevent age-related failures of SSCs important to safety, others seemed to be taking a passive or reactive stance. Differing maintenance philosophies and financial resources affect management's attention to

aging concerns. One utility considered its license renewal program to be founded on a strong maintenance program. None of the utilities had a separate or distinct program to address the management of aging. Most, if not all, appeared to rely on their maintenance programs to indirectly address aging.

The activities at every plant ensure that the infrastructure for understanding the aging problems exists in the operational and maintenance programs of the plant. Recent studies on nuclear plant aging, case studies on certain components and systems, and other related research activities both by the industry and the NRC have created an awareness among the utilities of aging of SSCs in nuclear power plants. This is evident from their adoption of such programs as Reliability Centered Maintenance and Life Cycle Management. Some plants have implemented advanced techniques to manage aging, which include vibration monitoring, thermography, Electronic Characterization and Diagnostics, and other testing and monitoring methods. Areas are noted for improvement with respect to management of aging in the present plant maintenance programs. In summary, after reviewing the 44 MTI reports, we believe that the process of adapting a forward-looking approach to the management of aging is in the initial stages.

Aging Management Program Insights

Preventive Maintenance. Preventive maintenance (PM) is the periodic, predictive, or planned maintenance of an SSC, which is performed before failure, to extend the service life by controlling degradation or failure. Every plant has a PM program as part of its plant maintenance program, specifically for those which are vital to plant safety and power generation. The PM program involves scheduled inspection activities for observing the equipment conditions; monitoring and surveillance testing of various equipment functional parameters; replacing degraded parts or parts with known life cycles; and performing routine maintenance activities, such as cleaning, repacking, and lubricating.

Strictly speaking, predictive maintenance is a form of PM performed continuously or at intervals governed by observed conditions to monitor, diagnose, or trend an SSC's functional or condition indicators. The results indicate current and future functional ability or the nature and schedule for planned maintenance. Examples are the scheduled in-service inspection and test required by the *American Society of Mechanical Engineers Boiler and Pressure Vessel Code* Section XI and plant Technical Specifications.

Several activities cited in the MTI reports suggest that the industry is striving to improve its existing PM programs. Most original PM program elements were developed in response to regulatory requirements, vendor recommendations, and good practices. Especially noteworthy were 13-week rolling maintenance schedules at a few plants in which an entire train of safety-related components is taken out of service for maintenance and surveillance testing. A Configuration Management Information System has been implemented at a few plants to enhance the effectiveness of the PM program. Other efforts included adopting improved testing methods, monitoring performance of the entire plant or certain systems in addition to individual equipment, trending of maintenance data, RCA, scram frequency analysis, and material condition management programs. Other analytical approaches included time series analysis of equipment failures, improved motor-operated valve (MOV) reliability, and aggressive resolution of immediate problems.

Many of the preceding activities were not focused to identify age-related deterioration occurring in the equipment. Rather, the PMs were performed to keep the component operable so as not to compromise plant availability. Some PM schedules were not implemented on a timely basis and, in fact, had items long overdue. In some cases, often without adequate justification, certain components, such as molded-case circuit breakers and instrument air system filters, were not subjected to PM for long intervals. Backlogs for PM were high at some plants.

As the benefits of a good PM program become evident, additional components are often added to the list for vibration monitoring, oil sampling, and periodic cycling. The PM frequencies chosen for particular equipment types were not uniform throughout the industry. The frequencies were usually based on good maintenance practices, vendor recommendations, component failure experience, outage planning, and management decisions regarding financial and staffing resources.

Predictive Maintenance and Condition Monitoring. Predictive maintenance and condition monitoring include diagnostic practices that can be useful to predict the remaining life to ensure the operational readiness until the next scheduled maintenance and to detect incipient degradation caused by aging effects. The most common practices include trending of degradation and failure rates, thermography, signature analysis of MOVs, and vibration analysis. From the available information, it was difficult to compare the predictive programs of one utility with those of another. Many utilities perform similar condition monitoring programs,

however, specifically valve surveillance testing using MOVATS (Motor-Operated Valve Analysis and Test System) or VOTES (Valve Operation Test Evaluation System), lubrication-oil analysis, vibration monitoring, and infrared imaging of intricate electrical circuits. Especially noteworthy was the microelectronic surveillance and calibration system at the Braidwood station to dynamically test instrument systems. We noted that use of advanced techniques is still in the early stages of implementation at most plants, but there was an increasing trend of use of such techniques.

The remaining life of equipment is assessed qualitatively on the basis of the information available from the EQ test or analysis programs, good maintenance practices in other industries, vendor recommendations, and operating experience. The Institute of Electrical and Electronics Engineers standards are used to predict the remaining life on the basis of the Arrhenius methodology. The overall performance of equipment is often characterized by the useful life of the weakest subcomponent. The Arrhenius methodology is a measure of chemical degradation of organic materials. Such materials are used typically for electrical insulation in cables, motors, transformers, and other such devices.

Postmaintenance Testing. Postmaintenance testing (PMT), as the name implies, is testing performed after maintenance to verify that the maintenance was performed correctly and that the SSC can function within its acceptance criteria. Such testing can also be a means for monitoring age-related degradation. PMT is sometimes referred to as operations verification testing, functional testing, channel checking, or time-delay testing, depending on the application to a particular component or a system. At times it is implemented at the next surveillance test. Otherwise it may involve inspection checking or just operating the equipment.

From our review of the MTI reports, we noted that in some cases, although these activities were very well documented and comprehensive in scope, acceptance criteria to confirm the operational readiness were very limited and vague. The manufacturers often were not able to define the thresholds that signify that the degree of degradation was unacceptable. In some cases, human-related problems were discovered during PMT, and appropriate actions were taken to restore the equipment conditions. Sometimes the PMT activity itself resulted in the need for corrective maintenance.

If the PMT does not identify ineffective maintenance, then aging can occur faster than expected. Thus the effectiveness of the maintenance can be measured by the

success of the PMT. We noted that documentation of PMT results was often poor. Although examples of well-documented and implemented PMT programs were cited in the reports, we concluded that PMT is an area that requires significant improvement at many plants.

Trending Analysis. Trending analysis is the evaluation of the statistical pattern of performance indicators over a period of time. These indicators are typically available from records of certain plant activities, including maintenance work requests and component-system functional or design parameters. Several computer-based software programs were used by almost all plants. Newer plants have an easier task to implement a trending program than the older facilities because they have the benefit of starting the data collection process in a trendable form early in plant life. Although a large number of plants seemed to have trending programs in place as part of their maintenance programs because of the inadequacies of the records and lack of commitment to trend the observed failures, we concluded that these programs are not adequate for understanding, detecting, and mitigating the effects of aging.

Root-Cause Analysis. Root-cause analysis is the in-depth evaluation of the causes and mechanisms of a failure event so that repetitive occurrence of this event can be prevented or minimized; thus maintenance backlogs and equipment outage time can be reduced. Inadequate analysis, inadequate support from the engineering support staff, insufficient information available for the RCA, and lack of commitment on the part of the management were among some of the deficiencies noted from this study. A couple of plants, in contrast, demonstrated cases of very well performed and documented RCA for battery chargers and main steam isolation valves. We concluded that RCA is an area not totally appreciated by all utilities and is an area that requires significant improvement.

Use of Probabilistic Risk Assessment. Probabilistic risk assessment has not been extensively used in the maintenance decisions. Only a few utilities use PRA for higher level decision making, such as scheduling system outages, justifying limiting conditions of operations, determining the importance of implementing modifications, and prioritizing their order of implementation. Use of PRA for maintenance decision making is still in the development stage.

SYSTEM-COMPONENT-LEVEL INSIGHTS

Several systems and components were chosen for this study to compare the kinds of maintenance practices

being performed at the nuclear power-plant facilities with the results and recommendations obtained from the NRC's aging studies. The systems chosen were auxiliary feedwater, feedwater, high-pressure injection, service water, and instrument air and emergency diesel generator air start systems and compressors. The components chosen were emergency diesel generators, electrical components (breakers, switchgear, relays and motor control centers), MOVs, and check valves.

This evaluation also provides a very useful alternative perspective. It yields a qualitative understanding of aging problems pertaining to specific systems or components. Examples of strengths and weaknesses in specific plant maintenance programs are also discussed. For the purpose of presentation, results for the service water system (SWS) and the check valves are discussed.

Service Water System

The SWSs perform vital safety functions as the final link between the reactor and the ultimate heat sink (i.e., river, lake, and cooling pond). On the basis of operating experience, the principal degradation mechanisms for SWS aging problems are corrosion, biofouling, and wear. NRC Generic Letter 89-13, concerning biofouling of safety-related equipment, and an Office for Analysis and Evaluation of Operational Data (AEOD) study have generated an awareness among the utilities of the problems associated with this system. Aging insights from the MTI reports include thinning of pipe walls caused by erosion-corrosion, resulting in through-wall leaks at the welded joints of carbon steel; absence of chemical treatment of spray ponds, resulting in valves and piping becoming filled with scale and sludge; chloride-induced stress corrosion; pump seal and packing leaks; accumulation of dirt at relay and switch contacts; and water hammer problems causing system vibration. Most utilities are aware of these problems as applicable to their plants. Nevertheless, we noted in the NRC reports instances of poor maintenance practices, lack of incorporation of industry-recommended practices, and inadequate RCA.

Because this system is a highly important, safety-related system, the components within the system are often subjected to a PM program, as is the case for other safety systems. However, the system uses raw water from an outside source, which is typically very harsh and causes deterioration of components faster than expected. Again, its continuous operating status during the plant's normal operation accelerates the aging process even further. Utilities are chemically treating the water to prevent corrosion and taking other preventive measures

Reactor Shutdown Experience

Compiled by J. W. Cletcher^a

This section presents a regular report of summary statistics relating to recent reactor shutdown experience. The information includes both numbers of events and rates of occurrence. It was compiled from data about operating events entered into the SCSS data system by the Nuclear Operations Analysis Center at the Oak Ridge National Laboratory. Whereas this compilation has hitherto covered three-month periods, the changes in publication schedule for *Nuclear Safety* and the delays caused by the changeover in funding have resulted in the tables in this issue covering the entire year 1993. Cumulative information, starting from May 1, 1984, is also shown. Updates on shutdown events included in earlier reports are excluded.

Table 1 lists information on shutdowns as a function of reactor power at the time of the shutdown for both

boiling-water reactors (BWRs) and pressurized-water reactors (PWRs). Only reactors in commercial operation at the start of the reporting period (January 1, 1993) are included. The second column for each reactor type shows the annualized shutdown rate for the reporting period. The third and fourth columns list cumulative data (numbers and rates) starting as of May 1, 1984.

Table 2 shows data on shutdowns by shutdown type: *Shutdowns required by Technical Specifications* are automatic scrams under circumstances where such a shutdown was required; *Intentional or required manual reactor protection system actuations* are manual shutdowns in which the operators, for reasons that appeared valid to them, took manual actions to actuate features of the reactor protection system; *Required automatic reactor protection system actuations* are actuations that the human

Table 1 Reactor Shutdowns by Reactor Type and Percent Power at Shutdown^a
(Period Covered is the Year 1993)

Reactor power (P), %	BWRs (37)				PWRs (75)			
	Number	Shutdown rate (annualized for period)	Cumulative number	Cumulative shutdown rate per reactor year ^b	Number	Shutdown rate (annualized for period)	Cumulative number	Cumulative shutdown rate per reactor year ^c
0	21	0.57	658	1.94	24	0.32	449	0.67
0 < P ≤ 10	5	0.14	125	0.37	4	0.05	162	0.24
10 < P ≤ 40	11	0.30	158	0.47	6	0.08	311	0.46
40 < P ≤ 70	8	0.22	146	0.43	2	0.03	167	0.25
70 < P ≤ 99	15	0.41	354	1.04	25	0.33	499	0.74
99 < P ≤ 100	35	0.95	451	1.33	73	0.97	1105	1.64
Total	95	2.57	1892	5.57	134	1.78	2693	4.00

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

^bBased on cumulative BWR operating experience of 339.45 reactor years.

^cBased on cumulative PWR operating experience of 672.87 reactor years.

^aOak Ridge National Laboratory.

Table 2 Reactor Shutdowns by Reactor Type and Shutdown Type^a
(Period Covered is the Year 1993)

Shutdown (SD) type	BWRs (37)				PWRs (75)			
	Number	Shutdown rate (annualized for period)	Cumulative number	Cumulative shutdown rate per reactor year ^b	Number	Shutdown rate (annualized for period)	Cumulative number	Cumulative shutdown rate per reactor year ^c
SDs required by Technical Specifications	14	0.38	245	0.72	14	0.19	387	0.58
Intentional or required manual reactor protec- tion system actuations	18	0.49	179	0.53	32	0.43	348	0.52
Required auto- matic reactor protection system actua- tions	47	1.27	887	2.61	75	0.97	1515	2.25
Unintentional or unrequired manual reactor protection sys- tem actuations	0	0.00	9	0.03	1	0.01	19	0.03
Unintentional or unrequired automatic reac- tor protection system actua- tions	16	0.43	572	1.69	12	0.16	424	0.63
Total	95	2.57	1892	5.57	134	1.76	2693	4.00

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

^bBased on cumulative BWR operating experience of 339.45 reactor years.

^cBased on cumulative PWR operating experience of 672.87 reactor years.

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

Table 3 lists information about shutdowns by reactor age category, both total numbers and rates in that category; it also shows cumulative results. Note that the age groups are not cohorts; rather reactors move into and out of the specified age groups as they age. The reactor age as used in this table is the number of full years between the start of commercial operation and the beginning of the reporting period (January 1, 1993, for this issue). The first line of this table gives the information for reactors licensed for full power but not yet in commercial operation on that date.

Table 3 Reactor Shutdowns by Reactor Type and Reactor Age^a
(Period Covered is the Year 1993)

Years in commercial operation (C.O.)	Exposure during the period (in reactor years)	BWRs (37)					PWRs (75)					
		Number		Shutdown rate (annualized for the period)	Cumulative number	Cumulative shutdown rate per reactor year	Exposure during the period (in reactor years)	Number		Shutdown rate (annualized for the period)	Cumulative number	Cumulative shutdown rate per reactor year
		Reactors	Shutdowns					Reactors	Shutdowns			
Not in C.O. ^b	1.000	1	0	0.00	330	23.60	0.326	1	2	6.14	336	34.24
First year of C.O.	0.000	0	0	0.00	121	9.00	0.413	1	2	4.84	278	10.07
Second through fourth year of C.O.	1.000	1	3	3.00	264	6.29	2.842	4	16	5.63	523	5.62
Fifth through seventh year of C.O.	5.503	7	11	2.00	171	4.50	11.069	15	18	1.63	310	3.36
Eighth through tenth year of C.O.	6.921	9	20	2.89	194	5.65	13.270	16	18	1.36	360	3.90
Eleventh through thirteenth year of C.O.	0.567	1	1	1.76	270	5.84	6.967	9	16	2.29	492	4.39
Fourteenth through sixteenth year of C.O.	1.369	3	2	1.46	395	6.23	5.665	9	11	1.94	362	3.30
Seventeenth through nineteenth year of C.O.	9.183	11	25	2.72	279	4.85	16.597	21	28	1.69	240	2.69
Twentieth through twenty-second year of C.O.	6.965	11	25	3.59	143	5.01	12.709	20	23	1.81	84	2.37
Twenty-third through twenty-fifth year of C.O.	4.468	6	8	1.79	39	3.82	2.823	4	0	0.00	21	1.77
Twenty-sixth through twenty-eighth year of C.O.	0.000	0	0	0.00	8	2.67	2.000	2	2	1.00	14	2.80
Twenty-ninth through thirty-first year of C.O.	1.000	1	0	0.00	8	2.90	0.000	0	0	0.00	5	1.67
Thirty-second through ninety-ninth year of C.O.	0.000	0	0	0.00	0	0.00	1.000	1	0	0.00	0	0.00
Total	37.974		95	2.50	2222	6.29	75.688		136	1.80	3025	4.43

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

^bThis category includes reactors licensed for full-power operation but not yet commercial. During this reporting period reactors in this category included 1 BWR (Shoreham) and 1 PWR (Comanche Peak 2).

Selected Safety-Related Events

Compiled by G. A. Murphy^a

ELECTRICAL TRANSIENT FOLLOWING THE LOS ANGELES EARTHQUAKE ON JANUARY 17, 1994^b

Summary

At 4:31 a.m. Pacific Standard Time (PST), Jan. 17, 1994, an earthquake measuring 6.6 on the Richter scale struck southern California and thus caused the western states power grid to separate. Transmission lines tripped and power plants tripped or ran back in Arizona, California, Colorado, Idaho, Montana, Nevada, New Mexico, Oregon, Utah, Washington, and Wyoming.^c

The Western Systems Coordinating Council (WSCC) bulk transmission system (the grid) separated into north and south islands. Wyoming, Utah, Colorado, New Mexico, El Paso (Texas), Arizona, southern Nevada, and parts of southern California and Mexico became the south island. British Columbia and Alberta (Canada), Washington, Oregon, Idaho, Montana, northern Nevada, and northern California became the north island. The frequency in the south island increased to a maximum of 60.8 Hz, whereas the frequency in the north island decreased to a minimum of 59.03 Hz and some loads were lost. A portion of southeastern Idaho was blacked out as well as Los Angeles, Burbank, and Glendale, California; parts of Portland, Oregon; and parts of Seattle, Washington.

About 45 transmission lines were reported to have tripped and 40 generating units tripped or ran back. Power was restored to these facilities in times ranging from 1 minute to several hours, whereas others were out of service for longer periods. Over 100,000 customers outside the quake area, mostly in Idaho, were without power for hours.

Diablo Canyon nuclear power station, in the north island, experienced a minimum frequency of 59.03 Hz and

a sustained frequency under 59.83 Hz for 20 minutes when the southern intertie, Midway-Vincent #1, #2, and #3, tripped. WNP 2 nuclear power station was also in the north island. Operating nuclear plants in the south island were San Onofre and Palo Verde.

The performance of the WSCC grid fell within the emergency operating criteria with the possible exception of the blackouts in Idaho. The estimated frequency for an earthquake-related loss of offsite power (LOOP) ranges from higher than that of sites with known grid reliability problems and low-to-moderate severe-weather hazards to higher than that of sites located in a high severe-weather hazard area, depending on the duration of the LOOP. Offsite power to a nuclear plant has not been lost because of the frequency swings, but the potential exists.

Discussion

Before the earthquake, an event occurred at Diablo Canyon that was to affect the response of that facility to the earthquake significantly. On Dec. 26, 1993, a static wire broke and the #11 500/230-kV transformer at the Midway substation tripped on sudden pressure. Midway-Kern 230-kV lines #1 and #2 and the Midway-Vincent 500-kV line #3 tripped (shown as "Lines involved in the December 26 event" in Fig. 1). Diablo Canyon 1 tripped from 100% power because of the line fault and a preexisting equipment problem. The excitation system isolation transducer #3 was out of calibration and sensed a nonexistent failure, which tripped the turbine erroneously. The grid frequency in the area dropped to 59.83 Hz and stabilized at 59.875 Hz. The reactor coolant pumps at Diablo Canyon 1 tripped on under-frequency. Twenty-one minutes later the frequency returned to 60 Hz, following the correction of an erroneous reading to a grid computer. The faulty transducer was replaced.

At 4:31 PST, Jan. 17, 1994, an earthquake that measured 6.6 on the Richter scale struck southern California. The epicenter was located in the San Fernando valley in the community of Northridge, a suburb of Los Angeles. The Los Angeles Department of Water and Power (LDWP) reported that all generating units in the basin tripped, and the Los Angeles area served by LDWP, Burbank Public Service Department, and the City of Glendale Public Service Department was blacked out.

^aOak Ridge National Laboratory.

^bCondensed from NRC AEOD Technical Review Report AEOD/T94-01, Mar. 16, 1994.

^cAll information concerning events that took place on Jan. 17, 1994, was obtained from the Department of Energy, Emergency Preparedness Office.

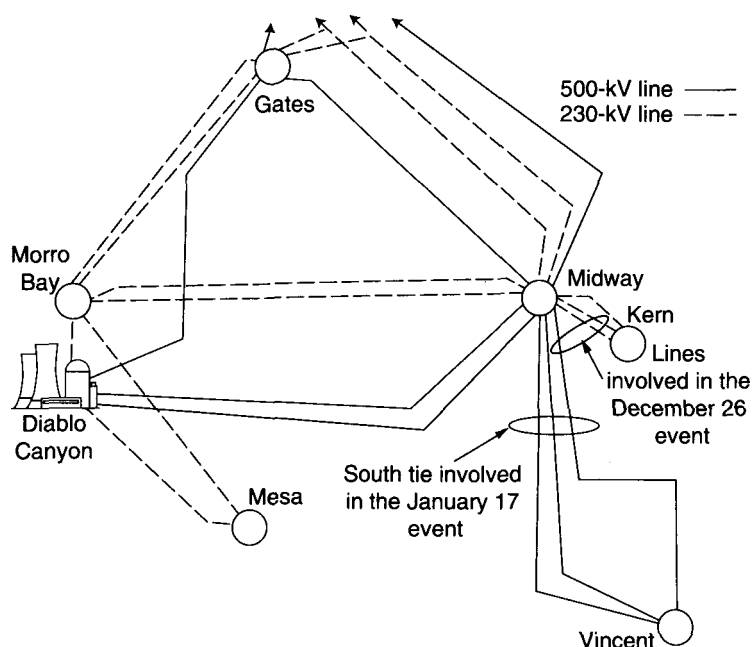


Fig. 1 Transmission map of the Diablo Canyon area.

The Nuclear Regulatory Commission (NRC) contacted the San Onofre and Diablo Canyon nuclear plants to determine what effects the earthquake had on their units. Control room personnel at San Onofre reported that they felt the shock but no motion indicators activated. Motion was neither felt by the Diablo Canyon personnel nor registered on the motion indicators. Both San Onofre and Diablo Canyon personnel reported frequency problems on the grid.^a Because of the frequency disturbances reported in the event notification and news reports of blackouts in Seattle, Washington, and Portland, Oregon, a call was made to the Emergency Preparedness Office of the Department of Energy (DOE) to determine the scope of the grid transient. DOE provided the NRC a copy of the Western States Coordinating Council preliminary report which showed a system-wide massive disturbance.

At the time of the earthquake, the southern intertie—three 500-kV lines, Midway–Vincent #1, #2, and #3—tripped and the Pacific D.C. intertie blocked. The resultant power surge flowed eastward and thus caused the Treasureton, Idaho, out-of-step scheme to activate. The grid in the western states began to separate. The Treasureton out-of-step scheme initiated the breakup of WSCC into islands. Southeastern Idaho separated and the 345/500-kV interties at the Jim Bridger plant in

Wyoming opened. Idaho power separated east of the Midpoint substation on three-phase faults, probably because of the out-of-step swing. Other lines tripped and thus separated Montana from Wyoming and Idaho. The grid within Utah began to separate and thus completed the formation of islands.

Wyoming, Utah, Colorado, New Mexico, El Paso (Texas), Arizona, southern Nevada, and parts of southern California and Mexico became the south island. British Columbia and Alberta (Canada), Washington, Oregon, Idaho, Montana, northern Nevada, and northern California became the north island. The frequency in the south island increased to a maximum of 60.8 Hz, whereas the frequency in the north island decreased to a minimum of 59.03 Hz, and some loads were lost. A portion of southeastern Idaho was blacked out as well as Los Angeles, Burbank, and Glendale, California; parts of Portland, Oregon; and parts of Seattle, Washington.

About 45 transmission lines were reported tripped, and 40 generating units tripped or ran back. Figure 2 illustrates the location of the earthquake, nuclear plants, tripped power plants, and blacked-out areas. These units produce about 6000 MW, or 4% of the total capacity of WSCC (including Canadian provinces and some northern Mexican areas). Power was restored to these facilities in times ranging from one minute to several hours, whereas others were out of service for extended periods. Over 100,000 customers in Idaho; Montana; Portland, Oregon;

^aEvent Notification EN 26627, dated Jan. 17, 1994, provided by the NRC Operations Center.

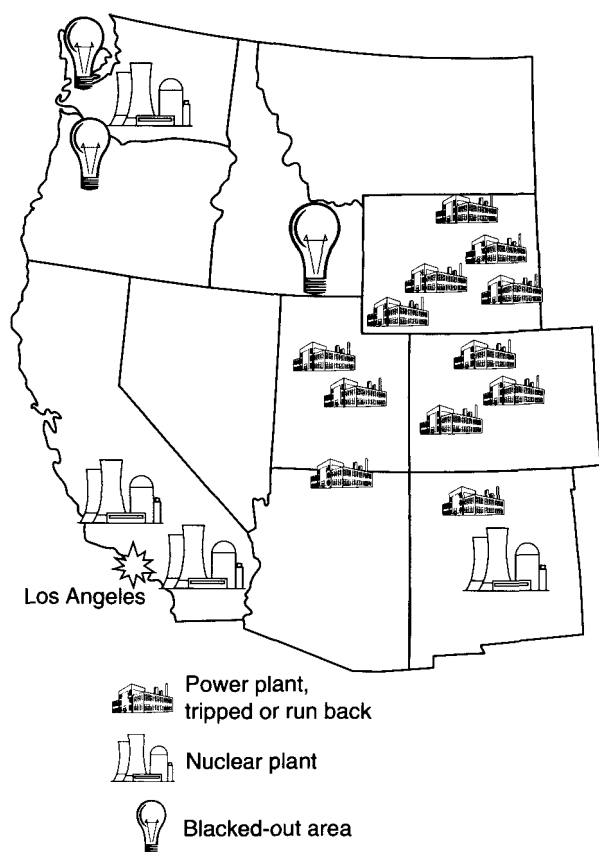


Fig. 2 Western Systems Coordinating Council (WSCC) map.

and Seattle, Washington—some several hundred miles from the Los Angeles quake area—were without power for hours.

Diablo Canyon, in the north island, experienced frequency of 59.03 Hz and a sustained frequency under 59.83 Hz for 20 minutes when the southern intertie, Midway-Vincent #1, #2 and #3, tripped (shown as “South tie involved in the January 17 event” in Fig. 1). WNP 2 was also in the north island. Operating nuclear plants in the south island were San Onofre and Palo Verde.

The initiating disturbance for the grid transient appeared to be the loss of the three 500-kV Midway-Vincent lines—a loss of three or more circuits on one right of way. In the “Criteria for Dynamic Performance of Interconnected Bulk Power Systems,” Section I Performance Levels,” four performance levels (A, B, C, and D) for the grid are defined. The initiating disturbance for this transient would cause a D-level of performance, which involves remedial actions that could include dropping of interruptible loads, tripping or runback of generators,

controlled opening of system interconnections, system islanding, automatic under-frequency load dropping, control direct dropping of firm load, sub-islanding, and generation separation. After the disturbance, transmission loads and substation voltages may be outside the emergency limits until they are readjusted. The “Basic Criteria” of the “General Operating Reliability Criteria” include the following statements: (1) “The bulk power systems will be operated at all times so that general system instability, uncontrolled separation, cascading outages, or voltage collapse will not occur as a result of the single most severe contingency.” (2) “Multiple contingency outages of a credible nature will be examined, and the system will be operated to protect against general system instability, uncontrolled separation or cascading outages for these contingencies.” (3) “Continuity of service is the primary objective of the Minimum Operating Criteria. Preservation of the interconnections during disturbances is a secondary objective except when preservation of interconnections will minimize the magnitude of load interruption or will expedite restoration of service to load.”

In the “Emergency Operating Procedures,” it is recognized that, regardless of many precautionary procedures, emergencies do occur. For load shedding and separating into islands, each WSCC member is required to determine separation points and islands and to initiate a program of automatic load shedding to arrest any frequency decay. This program would minimize the possibility of total grid collapse and prevent damage to equipment that grid collapse would cause. Island formation and load shedding would leave the system in a condition to rapidly restore loads and reestablish interconnections.

The initial under-frequency relays operate at 59.3 Hz; the next relays operate at 59.1 Hz. In areas that are isolated with excessive generation, automatic generator tripping or runback to prevent excessive over-frequency is warranted. In this event Intermountain units 1 and 2 tripped for this reason.

The utilities are required to “provide startup power to generating stations and off-site power to nuclear stations, where required.” In this event, no nuclear unit lost off-site power. Restoration is to be accomplished only when the systems conditions have recovered to the extent that lost loads can be restored without adverse effect.

Analysis

The Dec. 26, 1993, event uncovered the faulty transducer at Diablo Canyon 1, which, if left uncorrected,

^aAll information concerning the operating reliability criteria for WSCC was obtained from their 1992 EE-411 report to the Department of Energy, sections 5 and 6.

might have sensed the Midway-Vincent lines fault and caused Diablo Canyon to trip during the Jan. 17, 1994, earthquake. If the trip had occurred, the grid frequency might have fluctuated even more and thus increased the possibility of a LOOP at Diablo Canyon.

NUREG-1032, "Evaluation of Station Blackout Accidents at Nuclear Power Plants,"¹ discusses LOOPS at nuclear plants. It categorizes LOOPS as plant-centered, grid-related, and weather-related. Weather-related LOOPS were said to be influenced by plant location. Significant factors were (1) the reliability of the grid and (2) the likelihood of severe weather. Severe-weather-related grid disturbances were described as infrequent but may result in a longer duration LOOP. Events after the Los Angeles earthquake suggest a similarity between severe weather and earthquake-related LOOPS not caused by direct seismic effects. Although NUREG-1032 does address a seismic event causing a LOOP, it assumes a safe shutdown earthquake at the site occurring once in a thousand reactor years and recovery from it taking 8 to 24 hours. This 1994 earthquake occurred over a hundred miles away from a nuclear plant and did not cause a LOOP by damaging the transmission lines, but it did have the potential to cause a LOOP because of degraded frequency of the grid. The direct seismic effects of the earthquake to the plants were insignificant.

Figure 3.3 of NUREG-1032 plots the estimated frequency of a LOOP (per site year) versus the duration

(in hours) for five groups of plants called "Offsite Power Clusters." This figure is reproduced as Fig. 3, and Table 1 gives the definition of offsite power clusters.

The estimated frequency of a LOOP from an earthquake was calculated by assuming that the probability of a LOOP was 1 in 2, given the WSCC grid conditions following the earthquake, and that the plant had operated for 10 years. This gives a frequency of 0.05, which is

Table 1 Characteristics of Loss of Offsite Power Event Clusters

- 1 Sites with demonstrated high grid reliability and multiple sources of offsite power available through independent switchyard circuits and low severe-weather hazards or design features to limit loss of offsite power or hasten recovery from severe weather events.
- 2 Sites with demonstrated high grid reliability and low severe-weather hazards with design features to limit loss of offsite power or hasten recovery from severe weather events.
- 3 Sites located in moderate to high severe-weather hazard area and with limited design features to preclude loss of offsite power or hasten recovery from severe weather events.
- 4 Sites with known grid reliability problems and low to moderate severe-weather hazards or design features to limit loss of offsite power or hasten recovery from severe weather events.
- 5 Sites located in a high severe-weather hazard area and without design features to preclude loss of offsite power or hasten recovery from severe weather events.

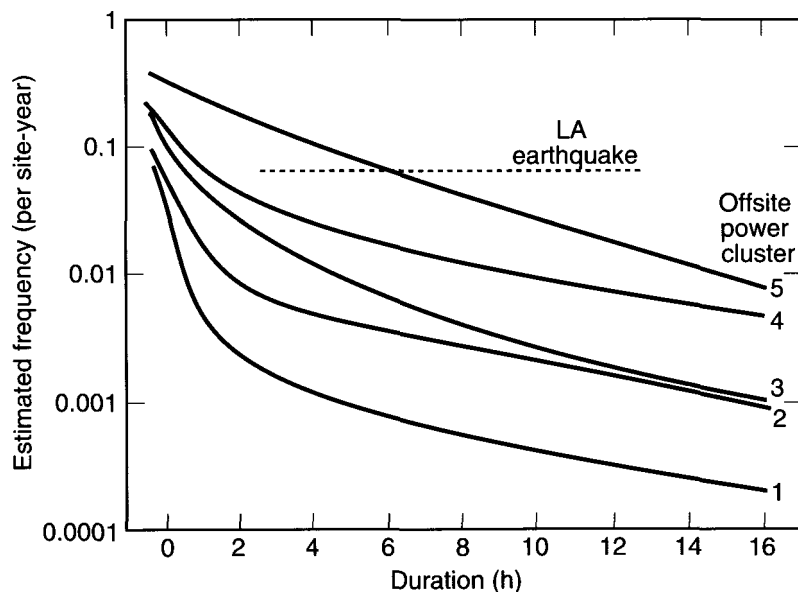


Fig. 3 Estimated frequency of a loss of offsite power (LOOP) event caused by an earthquake. The five numbered cases refer to the five groups of plants called "offsite power clusters" defined in Table 1.

shown as a dashed line in Figure 3 for a duration of 3 to 12 hours. Three hours is an estimate of the time to determine that the system is undamaged and to restore offsite power. Twelve hours is an estimate of the time to restore a damaged system and to restore offsite power.

This puts the estimated frequency for an earthquake-related LOOP exceeding specific durations in a range from higher than that of an offsite Power Cluster 4 to higher than that of an Offsite Power Cluster 5, depending upon the duration of the LOOP.

The earthquake precipitated a three line in one right-of-way fault, which caused the WSCC grid to lose load and separate to an island, to experience frequency perturbations and other problems within the D performance level predicted for this kind of occurrence. Outside the quake area, the system was rapidly restored in 1 to 3 hours.

All plant trips or runbacks and transmission-line trips occurred within 7 minutes or less, and at least 80% of the transmission-line trips occurred within less than 1 minute. There was no time for operator action to mitigate the circumstances. Should nuclear plants ever experience a loss of offsite power as a result of a natural disaster, WSCC has made the reestablishment of offsite power to the nuclear units a priority on the level with providing power to restart other generating units. This is reasonable because power must be generated to have offsite power available for the nuclear units.

Conclusions

1. The performance of the WSCC grid was within the emergency operating criteria with the possible exception of the blackouts in Idaho.

2. Events after the Los Angeles earthquake suggest a similarity between severe weather and earthquake-related LOOPS not caused by direct seismic effects. The estimated frequency for an earthquake-related LOOP ranges from higher than that of sites with known grid reliability problems and low to moderate severe-weather hazards to higher than that of sites located in a high severe-weather hazard area, depending on the duration of the LOOP.

3. No new issues involving nuclear plant safety were identified; however, the breadth and speed of the grid reaction to the initiating event should be recognized.

4. This kind of event is not limited to the WSCC because interactions between other Reliability Council member units can and do occur. The breadth of the reaction to the initiating event would depend upon its cause, where it occurred, time of year, time of day, and many other things; but an earthquake on the New Madrid fault in the Midwest should be expected to affect a large grid area of the United States and it could occur just as fast, with the potential to impact many more nuclear plants than the California earthquake because there are many more plants in the midwest and eastern United States.

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

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REGULATORY GUIDES

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

Table 1 Regulatory Guides

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

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

Division 1 Power Reactor Guides

- 1.009 (Revision 3) *Selection, Design, Qualification and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants*, July 1993.
- 1.084 (Revision 29) *Design and Fabrication Code Case Acceptability ASME Section III, Division 1*, July 1993.
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- DG-1025 (Draft guide, for comment) *Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence*, September 1993.

Division 3 Fuels and Materials Facilities Guides

- DG-3006 (Draft guide, for comment) *Standard Format and Content for Fire Protection Sections of License Applications for Fuel Cycle Facilities*, April 1993.
- DG-3008 (Draft guide, for comment) *Nuclear Criticality Safety Training*, January 1993.
- DG-3009 (Draft guide, for comment) *Topical Guidelines for Licensing Support System*, July 1993.

Division 8 Occupational Health Guides

- 8.009 (Revision 1) *Acceptable Concepts, Models Equations and Assumptions for Bioassay Program*, July 1993.
- 8.037 *ALARA Levels for Effluents from Materials Facilities*, July 1993.
- 8.038 *Control of Access to High and Very High Radiation Areas in Nuclear Power Plants*, June 1993.

Division 10 General Guides

DG-0003 (Draft guide) *Guide for Preparation of Applications for Licenses for Nonself-Contained Irradiators*, January 1994.

NUCLEAR STANDARDS

Standards pertaining to nuclear materials and facilities are prepared by many technical societies and organizations in the United States, including the Department of Energy (DOE) (NE Standards). When standards prepared by a technical society are submitted to the American National Standards Institute (ANSI) for consideration as an American National Standard, they are assigned ANSI standard numbers, although they may also contain the identification of the originating organization and be sold by that organization as well as by ANSI. We have undertaken to list here the most significant nuclear standards actions taken by organizations from January 1993 through March 1994. Actions listed include issuance for comments, approval by the ANSI Board of Standards Review (ANSI-BSR), and publication of the approved standard. Persons interested in obtaining copies of the standards should write to the issuing organizations.

American National Standards Institute

ANSI does not prepare standards; it is devoted to approving and disseminating standards prepared by technical organizations. However, it does publish standards, and such standards can be ordered from ANSI, Attention: Sales Department, 1430 Broadway, New York, NY 10018. Frequently, ANSI is an alternate source for standards also available from the preparing organization.

ANSI N14.6-1993 (Published) *Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More*, \$24.00.

ANSI N14.24-1985 (R1993, Reaffirmation) *Highway Route Controlled Quantities of Radioactive Materials—Domestic Barge Transport*.

ANSI N14.27-1986 (R1993, Reaffirmation) *Truckload Quantities of Radioactive Materials—Carrier and Shipper Responsibilities and Emergency Response Procedures for Highway Transportation Accidents*.

ANSI N42.13-1986 (R1993, Reaffirmation) *Calibration and Usage of "Dose Calibrator" Ionization Chambers for the Assay of Radionuclides*.

ANSI N42.16-1986 (R1993, Reaffirmation) *Sealed Radioactive Check Sources Used in Liquid Scintillation Counters*.

ANSI N320-1979 (R1993, Reaffirmation) *Performance Specifications for Reactor Emergency Radiological Monitoring Instrumentation*.

ANSI N323-1978 (R1993, Reaffirmation) *Radiation Protection Instrumentation Test and Calibration*.

ANSI/NFPA 801-1995 (Revision of ANSI/NFPA 801-1991, approved by ANSI/BSR) *Facilities Handling Radioactive Materials*.

ANSI Z244.1-1982 (R1993, Reaffirmation) *Personnel Protection—Lockout/Tagout of Energy Sources, Minimum Safety Requirements*.

ANSI Z400.1-1993 (Published) *Hazardous Industrial Chemicals—Material Safety Data Sheets—Preparation*, \$75.00.

BSR N14.2 (New standard, for comment) *Tiedowns for Truck Transport of Radioactive Material*.

BSR N15.36 (New standard, approved by ANSI/BSR) *Nondestructive Assay Measurements Control and Assurance*.

BSR N43.10-1984 (Reaffirmation of ANSI N43.10-1984, for comment) *Safe Design and Use of Panoramic, Wet Source Storage Gamma Irradiators*, \$12.00.

American Nuclear Society

Standards prepared by ANS can be obtained from ANS, Attention: Marilyn D. Weber, 555 North Kensington Avenue, LaGrange Park, IL 60525.

ANSI/ANS 15.11-1993 (Revision of ANSI/ANS 15.11-1987, approved by ANSI/BSR) *Radiation Protection at Research Facilities*.

ANSI/ANS 57.1-1992 (Published) *Design Requirements for Light Water Reactor Fuel Handling Systems*, \$55.00.

ANSI/ANS 58.14-1993 (New standard, approved by ANSI/BSR) *Safety and Pressure Integrity Classification Criteria for Light Water Reactors*.

ANSI/ANS 59.3-1992 (Published) *Nuclear Safety Criteria for Control Air Systems*, \$55.00.

BSR/ANS-3.2 (Revision of ANSI/ANS-3.2-1988, for comment) *Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants*, \$25.00.

BSR/ANS 8.6-1983 (R1988, for comment) *Safety in Conducting Subcritical Neutron-Multiplication Measurements in Situ*, \$14.00.

BSR/ANS 8.9-1987 (Reaffirmation of ANSI/ANS 8.9-1987, for comment) *Nuclear Criticality Safety Criteria for Steel-Pipe Intersections Containing Aqueous Solutions of Fissile Material*.

BSR/ANS 8.15-1981 (R1987, reaffirmation of ANSI/ANS 8.15-1981) *Nuclear Criticality Control of Special Actinide Elements*.

BSR/ANS 8.21 (New standard, approved by ANSI/BSR) *Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors*, \$10.00.

BSR/ANS 57.10 (Revision of ANSI/ANS 57.10-1984, for comment) *Design Criteria for Consolidation of LWR Spent Fuel*, \$10.00.

BSR/ANS 58.11 (Revision of ANSI/ANS 58.11-1983, for comment) *Design Criteria for Safe Shutdown Following Selected Design Basis Events in Light Water Reactors*, \$7.50.

American Society of Mechanical Engineers

Standards prepared by ASME can be obtained from ASME, Attention: R. D. Palumbo, 345 East 47th Street, New York, NY 10017.

ANSI/ASME AG-1b-1993 (Supplement to ANSI/ASME AG-1-1991, approved by ANSI/BSR) *Code on Nuclear Air and Gas Treatment*.

ANSI/ASME QME-1-1993 (New standard, approved by ANSI/BSR) *Appendix A to Section QR, Dynamic Qualification of Mechanical Equipment*.

BSR/ASME AG-1c (Addenda to ANSI/ASME AG-1-1991, for comment) *Code on Nuclear Air and Gas Treatment, Section RA—Refrigeration Equipment*, \$10.00.

BSR/ASME OMc (Supplement to ANSI/ASME OM Code-1990, for comment) *Code for Operation and Maintenance of Nuclear Power Plants*, \$35.00.

BSR/ASME OMc-S/G, Part 21 (Supplement to ANSI/ASME OM-S/G-1990, for comment) *Standards and Guides for Operation and Maintenance of Nuclear Power Plants, Part 21—Inservice Performance Testing of Heat Exchangers in LWR Plants*, \$39.00.

BSR/ASME QME-1 (Revision and redesignation of ANSI B16.41-1983, for comment) *Section QV, Functional Qualification Requirements for Active Assemblies*, \$30.00.

Institute of Electrical and Electronics Engineers

Standards prepared by IEEE can be obtained from IEEE, Attention: M. Lynch, 345 East 47th Street, New York, NY 10017.

ANSI/IEEE 1160-1993 (New standard, approved by ANSI/BSR) *Test Procedure for High-Purity Germanium Crystals for Radiation Detectors*.

ANSI/IEEE 1205-1993 (New standard, approved by ANSI/BSR) *Assessing, Monitoring, and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Stations*.

BSR/IEEE 338-1987 (Reaffirmation of ANSI/IEEE 338-1987, for comment) *Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Stations Safety Systems*, \$68.00.

BSR/IEEE 859-1987 (Reaffirmation of ANSI/IEEE 859-1987, for comment) *Terms for Reporting and Analyzing Outage Occurrences and Outage States of Electrical Transmission Facilities*, \$67.50.

BSR/IEEE/ANS 7-4.3.2 (Revision of ANSI/IEEE/ANS 7-4.3.2-1982, for comment) *Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations*, \$27.00.

International Standards

This section includes publications for any of the three types of international standards:

—IEC standards (International Electrotechnical Commission).

—ISO standards (International Standards Organization).

—KTA standards [Kerntechnischer Ausschuss (Nuclear Technology Commission)].

Standards originating from the IEC and ISO can be obtained from the American National Standards Institute (ANSI), International Sales Department, 1430 Broadway, New York, NY 10018.

The KTA standards are developed and approved by the Nuclear Safety Standards Commission (KTA). The KTA, formerly a component of the Gesellschaft für Reaktorsicherheit (GRS), is now integrated in the Federal Office for Radiation Protection (Bundesamt für Strahlenschutz BfS) in Salzgitter, Germany. Copies of these standards can be ordered from Dr. T. Kalinowski, KTA-Geschäftsstelle, Postfach 10 01 49, 3320 Salzgitter 1, Germany. These standards are in German, and, unless otherwise noted, an English translation is available from the KTA.

Prices for the international standards are shown in German currency (DM). The IEC, ISO, and KTA standards are all included in this issue.

IEC

IEC 45A(Central Office)137 (Draft standard, for comment) *Nuclear Reactors—Instrumentation and Control Systems Important for Safety—Instrumentation to Detect Leakage from Coolant Systems*, \$46.00.

IEC 476:1993 (Published) *Nuclear Instrumentation—Electrical Measuring Systems and Instruments Utilizing Ionizing Radiation Sources—General Aspects*, \$70.00.

IEC 1239:1993 (Published) *Nuclear Instrumentation—Portable Gamma Radiation Meters and Spectrometers Used for Prospecting—Definitions, Requirements and Calibration*, \$39.00.

IEC 1224:1993 (Published) *Nuclear Reactors—Response Time in Resistance Temperature Detectors (RTD)—In Situ Measurements*, \$30.00.

IEC 1225:1993 (Published) *Nuclear Power Plants—Instrumentation and Control Systems Important for Safety—Requirements for Electrical Supplies*, \$70.00.

IEC 1226:1993 (Published) *Nuclear Power Plants—Instrumentation and Control Systems Important for Safety—Classification*, \$70.00.

ISO

ISO 7195:1993 (Published) *Packaging of Uranium Hexafluoride (UF₆) for Transport*, \$72.00.

ISO 10981:1993 (Published) *Determination of Uranium in Reprocessing Plant Dissolver Solution—Liquid Chromatography Method*, \$25.00.

ISO/DIS 11932 (Draft standard, for comment) *Activity Measurements of Solid Materials Considered for Recycling, Re-Use or Disposal as Non-Radioactive Waste*, \$61.00.

ISO/DIS 12807 (Draft standard, for comment) *Leakage Testing on Packages for the Safe Transport of Radioactive Materials*.

KTA

KTA 1503.1 (Draft, revision of safety standard, 2/79) *Surveillance of the Discharge of Gaseous and Aerosol-Bound Radioactive Materials*, Issue 6/93.

KTA 3903 (Revision of safety standard, issue 11/82) *Testing and Operation of Lifting Equipment in Nuclear Power Plants*, Issue 6/93.

KTA 1504 (Revision of KTA 1504, issue 6/78) *Surveillance of the Discharge of Radioactive Materials in Liquid Effluents*, Issue 6/93.

KTA 3201.2 (Revision of KTA 3201.2, issue 3/84) *Components of the Reactor Coolant Pressure Boundary of Light Water Reactors: Part 2: Design and Analysis*, Issue 6/93.

Proposed Rule Changes as of Dec. 31, 1993^{a,b}

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

Number of part to be changed	Date published for comment	Date comment period expired	Date published; date effective	Topic or proposed effect	Current action and/or comment, <i>Federal Register</i> volumes and page numbers
10 CFR 0			2-3-93; 2-3-93	Conduct of employees; conforming amendments	Final rule in 58:7 (3825)
10 CFR 0			5-25-93; 6-24-93	Repeal of NRC Standards of Conduct regulations	Final rule in 58:99 (29951)
10 CFR 1	2-4-92	5-4-92	2-9-93; 7-1-93	Elimination of requirements marginal to safety	Published for comment in 57:23 (4166); final rule in 58:25 (7715)
10 CFR 1	2-24-92	3-6-92		Special review of NRC regulations	Published for comment in 57:36 (6299)
10 CFR 1	6-19-92	8-18-92; 9-30-92		Review of reactor licensee reporting requirements	Published for comment in 57:119 (27394); comment period extended in 57:153 (34886)
10 CFR 1 10 CFR 20 10 CFR 30 10 CFR 40 10 CFR 70 10 CFR 73			12-6-93; 12-13-93	NRC Region III telephone number and address change	Final rule in 58:232 (64110)
10 CFR 2	12-23-92	3-8-93		Availability of official records	Published for comment in 57:247 (61013)
10 CFR 2	3-17-93; 4-2-93	4-16-93; 5-3-93	3-17-93; 3-17-93; 4-2-93; 4-2-93	Policy and procedure for NRC enforcement actions; policy statement	Policy statement published for comment and adopted in 58:50 (14308); revision and extension of comment period in 58:62 (17321)
10 CFR 2 10 CFR 72	6-3-93	8-17-93; 10-1-93		Interim storage of spent fuel in an independent spent fuel storage installation; site-specific license to a qualified applicant	Published for comment in 58:105 (31478); comment period extended in 58:176 (48004)
10 CFR 2	9-29-93	11-15-93		Informal hearing procedures for materials licensing adjudications	Published for comment in 58:187 (50858)
10 CFR 2, 19, 20, 30, 31, 32, 34, 35, 36, 39, 40, 50, 61, 70			12-22-93; 1-1-94	Standards for protection against radiation; removal of expired material	Final rule in 58:244 (67657)
10 CFR 9			7-20-93; 7-20-93	Duplication fees	Final rule in 58:137 (38665)
10 CFR 12	8-2-93	9-1-93		Equal Access to Justice Act: implementation	Published for comment in 58:146 (41061)

Proposed Rule Changes as of Dec. 31, 1993 (Continued)

Number of part to be changed	Date published for comment	Date comment period expired	Date published; date effective	Topic or proposed effect	Current action and/or comment, <i>Federal Register</i> volumes and page numbers
10 CFR 19, 20, 21, 30, 36, 40, 51, 70, 170	12-4-90	3-4-91		Licenses and radiation safety requirements for large irradiators	Published for comment in 55:233 (50008)
10 CFR 19, 30, 40, 50, 60, 61, 70, 72, 150	6-15-93	7-15-93	10-8-93; 10-8-93	Whistleblower protection for nuclear power plant employees	Published for comment in 58:113 (33042); final rule in 58:194 (52406)
10 CFR 20 10 CFR 61	4-21-92	7-20-92		Low-level waste shipment manifest information and reporting	Published for comment in 57:77 (14500)
10 CFR 20			12-7-92; 1-6-93	Disposal of waste oil by incineration	Final rule in 57:235 (57649); corrections in 58:35 (11290)
19 CFR 20			12-8-92; 12-8-92	Revised standards for protection against radiation; minor amendments	Final rule in 57:236 (57877); correction in 58:249 (69219)
10 CFR 20	6-18-93	8-15-93; 9-20-93		Radiological criteria for decommissioning of NRC-licensed facilities; generic environmental impact statement (GEIS) for rulemaking, notice of intent to prepare a GEIS and to conduct a scoping process	Published for comment in 58:116 (33570); comment period extended in 58:154 (42882)
10 CFR 26 10 CFR 70 10 CFR 73	4-30-92	7-29-92	6-3-93; 11-30-93	Fitness-for-duty requirements for licensees who possess, use, or transport Category I material	Published for comment in 57:84 (18415); correction in 57:101 (22021); final rule in 58:105 (31467)
10 CFR 26	3-24-93	6-22-93		Modification of Fitness-for-Duty Program requirements	Published for comment in 58:55 (15810)
10 CFR 30 10 CFR 40 10 CFR 70 10 CFR 72	10-7-91	12-23-91	7-26-93; 10-25-93	Decommissioning recordkeeping and license termination: documentation	Published for comment in 56:194 (50524); final rule in 58:141 (39628)
10 CFR 30 10 CFR 40 10 CFR 70	2-20-92	4-30-92		Proposed method for regulating major materials licenses; availability of NUREG report	Published for comment in 57:34 (6077)
10 CFR 30, 40, 50, 70, 72	1-11-93	3-29-93	12-29-93; 1-28-94	Self-guarantee as an additional financial assurance mechanism	Published for comment in 58:6 (3515); final rule in 58:248 (68726)
10 CFR 30 10 CFR 40 10 CFR 70 10 CFR 72	1-13-93	3-29-93		Timeliness in decommissioning of materials facilities	Published for comment in 58:8 (4099)

(Table continues on the next page.)

Proposed Rule Changes as of Dec. 31, 1993 (Continued)

Number of part to be changed	Date published for comment	Date comment period expired	Date published; date effective	Topic or proposed effect	Current action and/or comment, <i>Federal Register</i> volumes and page numbers
10 CFR 30 10 CFR 40 10 CFR 50 10 CFR 70 10 CFR 72	2-2-93	4-5-93		Procedures and criteria for on-site storage of low-level radioactive waste	Published for comment in 58:20 (6730)
10 CFR 30 10 CFR 35	5-6-93	8-23-93; 12-31-94	7-22-93; 8-23-93	Authorization to prepare radiopharmaceutical reagent kits and elute radiopharmaceuticals for therapy; extension of expiration date	Extension of expiration date in 58:86 (26938); final rule and extension of expiration of interim rule in 58:139 (39130)
10 CFR 30 10 CFR 32 10 CFR 35	6-17-93	10-15-93		Preparation, transfer for commercial distribution, and use of byproduct material for medical use	Published for comment in 58:115 (33396)
10 CFR 31 10 CFR 32	12-27-91	3-12-92		Requirements for the possession of industrial devices containing byproduct material	Published for comment in 56:248 (67011)
10 CFR 31 10 CFR 32	11-27-92	3-29-93		Requirements concerning the accessible air gap for generally licensed devices	Published for comment in 57:229 (56287)
10 CFR 40	10-28-92	1-26-93		Licensing of source material	Published for comment in 57:209 (48749)
10 CFR 40 10 CFR 72 10 CFR 74 10 CFR 75 10 CFR 150	1-26-93	4-26-93		Licensee submittal of data in computer-readable form	Published for comment in 58:15 (6098)
10 CFR 40	11-3-93	12-17-93		Uranium mill tailings regulations; conforming NRC requirements to EPA standards	Published for comment in 58:211 (58657)
10 CFR 50 10 CFR 52	1-7-92	3-9-92	4-26-93; 5-26-93	Training and qualification of nuclear power plant personnel	Published for comment in 57:4 (537); final rule in 58:78 (21904); corrections in 58:138 (39092)
10 CFR 50	4-21-92	7-6-92		Loss of all alternating current power	Published for comment in 57:77 (14514); withdrawn in 58:133 (37884)
10 CFR 50	9-28-92	12-28-92		Acceptability of plant performance for severe accidents; scope of consideration in safety regulations	Published for comment in 57:188 (44513)

Proposed Rule Changes as of Dec. 31, 1993 (Continued)

Number of part to be changed	Date published for comment	Date comment period expired	Date published; date effective	Topic or proposed effect	Current action and/or comment, <i>Federal Register</i> volumes and page numbers
10 CFR 50 10 CFR 52 10 CFR 100	10-20-92	2-17-93; 3-24-93; 6-1-93		Reactor site criteria, including seismic and earthquake engineering criteria for nuclear power plants and proposed denial of petition for rulemaking from Free Environment, Inc., et al.	Published for comment in 57:203 (47802); comment period extended in 58:2 (271); extended again in 58:57 (16377)
10 CFR 50	3-22-93	5-6-93	6-23-93; 7-10-96	Monitoring the effectiveness of maintenance at nuclear power plants	Published for comment in 58:53 (15303); final rule in 58:119 (33993)
10 CFR 50 10 CFR 54	5-14-93	6-14-93	8-27-93; 9-27-93	FSAR update transmittals	Published for comment in 58:92 (28523); final rule in 58:165 (45243)
10 CFR 50	6-28-93	9-13-93		Production and utilization facilities; emergency planning and preparedness-exercise requirements	Published for comment in 58:122 (34539)
10 CFR 50	6-30-93	9-13-93		Notification of spent fuel management and funding plans by licensees of prematurely shut down power reactors	Published for comment in 58:124 (34947)
10 CFR 50			7-22-93; 7-22-93	Final policy statement on Technical Specification improvements for nuclear power plants	Final policy statement in 58:139 (39132)
10 CFR 51	9-17-91	12-16-91; 3-16-92		Environmental review for renewal of operating licenses	Published for comment in 56:180 (47016); comment period extended in 56:228 (59898)
10 CFR 52	11-3-93	1-3-94		Rulemakings to grant standard design certification for evolutionary light water reactor designs	Advance notice of proposed rulemaking published in 58:211 (58664)
10 CFR 52			12-30-93; 1-22-93	Combined licenses; conforming amendments; response to post-promulgation comment	Post-adoption comment published in 58:249 (69220)
10 CFR 55	5-20-93	7-19-93		Operator's licenses	Published for comment in 58:96 (29366)
10 CFR 60	7-9-93	10-7-93		Disposal of high-level radioactive wastes in geologic repositories; investigation and evaluation of potentially adverse conditions	Published for comment in 58:130 (36902)
10 CFR 61	3-6-92	4-6-92	6-22-93; 7-22-93	Licensing requirements for land disposal of radioactive wastes	Published for comment in 57:45 (8093); final rule in 58:118 (33886)

(Table continues on the next page.)

Proposed Rule Changes as of Dec. 31, 1993 (Continued)

Number of part to be changed	Date published for comment	Date comment period expired	Date published; date effective	Topic or proposed effect	Current action and/or comment, <i>Federal Register</i> volumes and page numbers
10 CFR 72	6-26-92	9-9-92; 1-21-93; 5-17-93	4-7-93; 5-7-93; 10-5-93; 11-4-93	List of approved spent fuel storage casks: additions	Published for comment in 57:124 (28645); comment period extended in 58:12 (5301); final rule in 58:65 (17848); comment period extended in 58:72 (19786); final rule in 58:191 (51762)
10 CFR 72	5-24-93	8-9-93; 11-9-93		Emergency planning licensing requirements for independent spent fuel facilities (ISFSI) and monitored retrievable storage facilities (MRS)	Published for comment in 58:98 (29795); comment period extended in 55:166 (45463)
10 CFR 72	9-14-93	11-29-93		Notification of events at independent spent fuel storage installations and the Monitored Retrievable Storage installation	Published for comment in 58:176 (48004)
10 CFR 73	12-13-91	3-13-92	8-31-93; 2-28-93(?)	Physical fitness programs and day firing qualifications for security personnel at Category 1 license fuel cycle facilities	Published for comment in 56:240 (65024); final rule in 58:167 (45781); corrections in 58:177 (48424)
10 CFR 73	5-29-92	8-12-92	3-15-93; 4-14-93	Clarification of physical protection requirements at fixed sites	Published for comment in 57:104 (22670); final rule in 58:48 (13699)
10 CFR 73 10 CFR 74			5-21-93; 6-21-93	Licenses' announcements of safeguards inspections	Final rule in 58:97 (29521)
10 CFR 73	10-6-93	12-20-93		Annual physical fitness performance training for tactical response team members, armed response personnel, and guards at Category 1 licensees	Published for comment in 58:192 (52035)
10 CFR 73	11-4-93	1-3-94		Protection against malevolent use of vehicles at nuclear power plants	Published for comment in 58:212 (58804); correction in 58:217 (59965)
10 CFR 110	2-7-90	3-9-90		Import and export of radioactive wastes	Advance notice of proposed rulemaking for comment in 55:26 (4181); corrections in 55:57 (10786);
	4-28-92	7-13-92			published for comment in 57:82 (17859)
10 CFR 110			3-9-93; 3-9-93	Export and import of nuclear equipment and material; clarifying amendments	Final rule in 58:44 (12999)
10 CFR 110	3-17-93	4-16-93		Specific licensing of exports of certain alpha-emitting radio-nuclides and byproduct material	Published for comment in 58:50 (14344)

Proposed Rule Changes as of Dec. 31, 1993 (Continued)

Number of part to be changed	Date published for comment	Date comment period expired	Date published; date effective	Topic or proposed effect	Current action and/or comment, <i>Federal Register</i> volumes and page numbers
10 CFR 110	10-28-93	1-10-94	10-28-93; 11-29-93	Export and import of nuclear equipment and material; export of high-enriched uranium	Final rule in 58:207 (57962)
10 CFR 140			8-12-93; 8-20-93	Adjustment of the maximum standard deferred premium	Final rule in 58:154 (42851)
10 CFR 170 10 CFR 171	4-19-93	7-19-93		NRC fee policy; request for public comment	Published for comment in 58:73 (21116)
10 CFR 170 10 CFR 171	4-23-93	5-24-93; 8-18-93		FY 1991 and 1992 proposed rule implementing the U.S. Court of Appeals decision and revision of fee schedule; 100% fee recovery, FY 1993	Published for comment in 58:77 (21662); corrections in 58:93 (28801) and in 58:96 (29454); final rule in 58:137 (38666); comment period extended in 58:139 (39174); corrections in 58:166 (45553)
10 CFR 171	9-29-93	10-29-93		Restoration of the generic exemption from annual fees for non-profit educational institutions	Published for comment in 58:187 (50859)
48 CFR 20	10-2-89	12-1-89		Acquisition regulation (NRCAR)	Published for comment in 54:189 (40420); corrections in 58:43 (12988)
48 CFR 2012 48 CFR 2015 48 CFR 2030 48 CFR 2052			5-3-93; 5-3-93; 9-8-93; 9-8-93	NRC Acquisition Regulation; minor amendments	Final rule in 58:83 (26253); revised final rule in 58:172 (47220)

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

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

The Authors

Chernobyl Accident Management Actions

Alexander Roman Sich holds the B.S. degree in nuclear engineering from Rensselaer Polytechnic Institute and the M.A. degree in Soviet studies from Harvard University. He completed his Ph.D. degree in nuclear engineering at the Massachusetts Institute of Technology (MIT) in January 1994. His thesis at MIT was a broad reappraisal of the Chernobyl accident and its consequences. He spent a year and a half living in the town of Chernobyl as the first westerner permitted to work closely with the members of the Chernobyl Complex Expedition—the small group of Russian and Ukrainian scientists studying the remains of Unit 4. The conclusions he reached in his thesis confirm earlier suspicions by western experts that more radioactivity was released as a result of the accident than claimed by the Soviets.

The IAEA-ASSET Approach to Avoiding Accidents is to Recognize the Precursors to Prevent Incidents

Frigyes Reisch is Chief Engineer at the Swedish Nuclear Power Inspectorate (SKI). He recently returned from an extended service at the International Atomic Energy Agency (IAEA). In the framework of different IAEA activities, he spent periods as leader and member of international working groups at all the RBMK reactors (Kursk/Russia, Leningrad/St. Petersburg/Russia, Ignalina/Lithuania, Smolensk/Russia, Chernobyl/Ukraine) and also at VVER reactors (Paks/Hungary, Dukovany/Czech Republic, Rovno/Ukraine) and visited also Lovisa/Finland. He is the leader of a working group at the International Electrotechnical Commission (IEC) preparing I&C standards and also for a joint IAEA-IEC working team for improving RBMK safety, instrumentation. Before his IAEA service, he was the head of SKI's inspection department. At the two major crises, at the detection of the Chernobyl accident in 1986 and the Swedish grid collapse in 1983, he was in charge of

conducting the activities of the Inspectorate. Before joining SKI he worked for Studsvik and before that for ABB. He has the Techn. lic. degree (international equivalent to Ph.D.) from the Physics faculty of Stockholm's Royal Institute of Technology and the Dipl. Ing. degree (international equivalent to M. Sc.) from the Electrical Engineering faculty of Budapest's Technical University. He has authored several articles on nuclear safety subjects for *Nuclear Safety* and other publications.

A Review of the Available Information on the Triggering Stage of a Steam Explosion

David F. Fletcher received his Ph.D. degree from the University of Exeter in the United Kingdom (UK) in 1982 for research into the computation of heat and mass transfer in separated flows. He then joined the United Kingdom Atomic Energy Authority (now called AEA Technology), working at their Winfrith and Culham Laboratories for a total of 10 years. Most of this time he worked on the study of steam explosions and the modeling of single and multiphase flows. He played a key role in the production of the Probabilistic Safety Assessment (PSA) for alpha-mode failure for the first UK pressurized-water reactor (Sizewell B). In 1992 he moved to the University of Sydney, where he teaches fluid dynamics and heat transfer in the Mechanical Engineering Department. He has continued his research work on steam explosions and is attempting to apply this work in the process industry.

Analysis and Modeling of Flow-Blockage-Induced Steam Explosion Events in the High-Flux Isotope Reactor

R. P. Taleyarkhan, V. Georgevich, C. W. Nestor, U. Gat, B. L. Lepard, D. H. Cook, J. Freels, S. J. Chang, C. Luttrell, R. C. Gwaltney, and J. Kirkpatrick. Current address: Oak Ridge National Laboratory, Oak Ridge, TN 37831.

An Analysis of Disassembling the Radial Reflector of a Thermionic Space Nuclear Reactor Power System

Mohamed S. El-Genk is a professor of chemical and nuclear engineering and the Director of the Institute for Space Nuclear Power Studies at the University of New Mexico. He received the B.Sc. degree in 1968 and the M.S. degree in 1975, both in nuclear engineering from the University of Alexandria, Egypt. He received the Ph.D. degree in nuclear engineering from the University of New Mexico (UNM) in 1978. Before joining UNM in 1981, he worked in industry for 13 years. He has published extensively in the areas of light-water reactor and liquid-metal fast breeder reactor safety and thermal-hydraulics, severe accident analyses and steam explosion, boiling and convective heat transfer, nuclear fuel behavior, and space nuclear power and propulsion. His current research interests also include pool boiling from inclined and downward facing surfaces; natural and combined convection in rod bundles; heat pipes; cooling of heated surfaces with swirling air jets; thermionic conversion; and design, modeling, and safety of space nuclear power and propulsion systems.

Dmitry Paramonov is an M.S. degree student in the Department of Chemical and Nuclear Engineering and a research assistant in the Institute for Space Nuclear Power Studies at the University of New Mexico. He received a degree in mechanical engineering, Power and Propulsion Systems of Spacecrafts, from Moscow Aviation Institute, Russia, in 1991. His current research includes modeling and design of space nuclear power systems and thermionic energy conversion.

Standards for High-Integrity Software

Dolores R. Wallace leads the High Integrity Software Systems Assurance project at the Computer Systems Laboratory of the National Institute of Standards and Technology. She is responsible for research, development of standards and guidelines, and technology transfer for Federal agencies and industry for the assurance of software in high integrity systems. She has served as Co-Chair of the September 1993 Digital Systems Reliability and Nuclear Safety Workshop, Chair of the Computer Assurance (COMPASS) Board of Directors, and a member of the Quality Assurance Institute Advisory Board. She has been an Institute of Electrical

and Electronics Engineers (IEEE) lecturer on standards and is active in international and national standards organizations. She was guest editor of the *IEEE Software* issue on software verification and validation. She received the M.S. degree in mathematics from Case Western University in Cleveland, Ohio. Current address: National Institute of Standards and Technology, Gaithersburg, MD 20899.

D. Richard Kuhn is a computer scientist in the Computer Systems Laboratory at the National Institute of Standards and Technology (NIST), where he is responsible for analysis techniques and formal method applications in computer security and open systems. He has published more than 25 papers on this work and received a Bronze medal from NIST in 1990 for contributions to open system standards. Before joining NIST in 1984, Kuhn worked as a systems analyst with NCR Corporation and the Johns Hopkins University Applied Physics Laboratory. He received the M.S. degree in computer science from the University of Maryland and the M.B.A. degree from the College of William and Mary. Current address: National Institute of Standards and Technology, Gaithersburg, MD 20899.

Laura M. Ippolito is a member of the High Integrity Software Systems Assurance project in the Computer Systems Laboratory (CSL) of the National Institute of Standards and Technology. She is responsible for research, development of standards and guidelines, and technology transfer of software engineering processes, methods, and techniques. She serves as Chair of Local Arrangements of the COMPASS (Computer Assurance) conferences and coordinates the CSL's Lecture Series on High Integrity Software Systems. Ippolito participates in the Working Group for the Institute of Electrical and Electronics Engineers Standard 1028 on audits and reviews. She received the B.A. degree in mathematics from Hood College and is a member of Pi Mu Epsilon mathematical honor society. Current address: National Institute of Standards and Technology, Gaithersburg, MD 20899.

Leo Beltracchi is a member of the research staff at the U.S. Nuclear Regulatory Commission (NRC). His current projects are directed toward the development of the technical basis for regulations on software for nuclear power plant safety systems. He was also honored as the NRC's 1994 Federal Engineer of the Year. He has also served as a project manager for planning, organizing, and coordinating major research activities dealing with human factors in nuclear safety.

Adoption of New Design Features for the Next Generation Nuclear Power Reactors

L. S. Tong received the Ph.D. degree in mechanical engineering at Stanford University in 1956. He worked on thermal-hydraulic design and development of early Westinghouse commercial pressurized-water reactors (PWRs) for 17 years and was given the Westinghouse Order of Merit award for his major contribution in new thermal designs of PWRs. He then served the Nuclear Regulatory Commission (NRC) in administering light-water-reactor (LWR) safety research programs for 10 years and received a Meritorious Service Award at NRC for his success in developing the overall logic of LWR safety research programs. Since 1983 he has served as an independent consultant on reactor safety and design for both the domestic and international nuclear power industries.

Review of Nuclear Piping Seismic Design Requirements

Gerry Slagis is a registered professional mechanical engineer with a consulting practice, GC Slagis Associates, located in California. He has a B.S. in mechanical engineering from the University of Detroit and an M.S. in mechanical engineering from the University of Southern California. He has over thirty years engineering experience in the aerospace and nuclear fields. His nuclear experience includes high-temperature code analyses on FFTF test facilities and B31 and Section III code analyses on conventional nuclear power plants. He is a national authority on piping design, seismic engineering, and the Section III nuclear code. For the past ten years, he has been chairman of the ASME Section III Working Group on Piping Design.

Samuel E. Moore is a senior development specialist in the Engineering Technology Division at the Oak Ridge National Laboratory. He has a B.S. in mechanical engineering from the University of New Mexico and an M.S. in the same discipline from the University of Tennessee. Moore has 39 years of experience in structural analysis, structural mechanics research, development of design criteria, and project management for both experimental and commercial nuclear power plants. He has published 57 related papers and reports. He is a member of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Committee (BPVC)

and the Pressure Vessel Research Council (PVRC). He is past chairman of the PVRC Subcommittee on Piping, Pumps, and Valves and immediate past chairman of the PVRC Committee on Piping and Nozzles. He is also a member of the ASME-BPVC Working Group on Piping Design and the Subgroup on Design for Section III of the *ASME Boiler and Pressure Vessel Code*.

PC-Based Probabilistic Safety Assessment Study for a Geological Waste Repository Placed in a Bedded Salt Formation

Sajjad Ali Khan is a senior safety and environmental consultant in SAK Engineering, Islamabad, Pakistan. He received the B.Engg (Mining) degree in 1979 and the M.Sc (Nucl. Engg) degree in 1982. His main field of interest has been safety in potentially hazardous industries (for example, chemical, mining, and nuclear), especially in developing countries. Having 14 years of experience in safety analysis and design evaluation, he is also a registered consultant with World Bank, Asian Development Bank, UNIDO, UNDP, and ACIL Australia. He has consulted in a variety of industrial situations on safety and environmental issues and specially those related to atmospheric dispersion of hazardous pollutants, waste management, risk and reliability analysis, and design evaluation. Currently he is working on aspects of risk and safety analysis for various waste disposal options in hazardous industries in geological rock formations.

Managing Aging in Nuclear Power Plants: Insights from NRC's Maintenance Team Inspection Reports

Anthony Fresco received the M.S. degree in mechanical engineering from the Polytechnic Institute of New York in 1974. He is a research engineer in the Department of Advanced Technology at Brookhaven National Laboratory; he has had more than 20 years experience in the power industry, including utility and architect engineering experience. More recently, he has been involved in developing probabilistic risk assessment-based guidance for nuclear plant inspection, has participated in numerous inspections of nuclear plants for postfire safe shutdown capability under

10 CFR 50 Appendix R, and has performed several evaluations of utility in-service testing programs for pumps and valves as required by the ASME Code, Section XI. Current address: Brookhaven National Laboratory, Department of Advanced Technology, Engineering Technology Division, Upton, New York 11973.

Mano Subudhi received the Ph.D. degree in mechanical engineering from the Polytechnic Institute of

New York in 1974. He is an engineering scientist in the Department of Advanced Technology and a member of the scientific staff at Brookhaven National Laboratory. For the last 20 years, his research interests have been in the area of structural analysis, seismic design, and aging of nuclear plant components and systems. Current address: Brookhaven National Laboratory, Department of Advanced Technology, Engineering Technology Division, Upton, New York 11973.

SYMPOSIUM ON RADIOACTIVE AND MIXED WASTE—RISK AS A BASIS FOR WASTE CLASSIFICATION

Las Vegas, Nevada, November 9, 1994

This symposium is sponsored by the National Council on Radiation Protection and Measurements (NCRP).

For additional information, contact Mr. William M. Beckner, Deputy Executive Director, NCRP, 7910 Woodmont Ave., Suite 800, Bethesda, MD 20814-3095. Telephone: (301) 657-2652. Fax: (301) 907-8768.

1995 INCINERATION CONFERENCE

Seattle, Washington, May 8–12, 1995

This conference is sponsored by the University of California, Irvine, with expected continued co-sponsorship by the U.S. Department of Energy, the U.S. Environmental Protection Agency, the American Society of Mechanical Engineers (ASME), the American Institute of Chemical Engineers (AIChE), the Air & Waste Management Association, the American Nuclear Society (ANS), the Coalition for Responsible Waste Incineration, and the Health Physics Society.

This world renowned symposium covers thermal treatment technologies for the management of special waste streams: Radioactive, hazardous chemical, mixed, chemical/pharmaceutical, explosive, and chemical munitions.

The technical program will consist of contributed and invited papers on topics of current interest to waste-management professionals, including research and development programs, operating thermal treatment system experience, design considerations, programmatic issues, and other issues of interest to professionals currently using or considering the use of thermal treatment technologies.

Three pre-conference or post-conference courses will be offered: Fundamentals, Tutorials, and Medical Waste. An exhibit of products and services will also be featured.

For additional information, contact Ms. Lori Barnow, University of California, Office of Environment, Health & Safety, Irvine, Calif., 92717-2725 U.S.A. Telephone: (714) 856-7066. Fax: (714) 856-8539.

NINTH POWER PLANT DYNAMICS CONTROL AND TESTING SYMPOSIUM

Knoxville, Tenn., May 24–26, 1995

This symposium is sponsored by the University of Tennessee College of Engineering, Department of Nuclear Engineering, and the Measurement and Control Engineering Center. It is intended to be a forum for engineers and researchers from electric utilities, power plant equipment manufacturers, research organizations, and universities to consider current and future problems in dynamic modeling, control, diagnostics and maintenance of power generating stations (nuclear, fossil, and alternative sources).

The symposium will cover the following topics: power plant modeling and simulation, power plant control, plant testing, training simulators, thermal performance monitoring, digital upgrading of instrumentation and control and safety systems in nuclear power plants, applied artificial intelligence for modeling and control, diagnostics and prognosis for plant components, advances in sensors, measurement systems, data analysis and signal validation, plant information systems and integration, methods for predictive maintenance and plant life extension, and automation of diagnostics using nondestructive examination methods.

For additional information, contact Prof. Belle R. Upadhyaya, Nuclear Engineering Department, Pasqua Engineering Building, The University of Tennessee, 1004 Estabrook Rd., Knoxville, TN 37996-2300. Telephone: (615) 5048. Fax: (615) 974-0668.

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