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

- event assessments
- performance indicators
- data systems development
- trends and patterns analyses
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Editorial Staff

The Chornobyl Accident

Edited by M. D. Muhlheim

The Chornobyl Accident Revisited, Part III: Chornobyl Source Term Release Dynamics and Reconstruction of Events During the Active Phase

By A. R. Sich[a]

[Editor's Note: The transliteration from Ukrainian to English is Chornobyl, whereas the Russian transliteration is Chernobyl.]

Abstract: *Chornobyl radioisotope release data presented by the Soviets at Vienna in August 1986 are reviewed and compared with newly available release data for the period of the active phase (t = 0+ up to 10 days). An analysis of these data indicates that radioisotopes were released under roughly isothermal conditions. Moreover, the releases of 17 isotopes analyzed are surprisingly close in magnitude, both with respect to their normalized mass releases and with respect to their release efficacies relative to Zr-95. On the basis of the information presented in this and the previous two articles of this series, a sequence of events is postulated as to what may have occurred to the Unit 4 core during the active phase. This scenario strongly contradicts accounts based on information presented by the Soviets in Vienna in August 1986. The release of eight volatile radioisotopes is estimated to be 92 MCi. This is substantially more than the total release of 50 MCi (excluding noble gases) claimed by the Soviets and confirms western suspicions that more was released.*

From the early development of nuclear reactors, radioactive fission products as well as actinides generated in the fuel during operation were recognized as the major biological hazard that had to be properly contained. For purposes of analyses, these fission products are usually divided into volatility groups (for example, although gases under ambient conditions and therefore of extremely high volatility, the noble gas fission products krypton and xenon are not considered strong biological hazards because of their relatively low fission yield, high escape probability, and chemical inertness). Conversely, iodine, cesium, and (to some extent) tellurium are considered to be the most important fission products in the *early* stages of a severe accident because they exhibit similar high volatilities and diffusion properties. Although I-131 has a relatively short half-life (8.04 d), it is a particularly hazardous fission product, and therefore its chemical forms and behavior after releases are important. Cesium, whose fission yield is approximately twice that of iodine, has a longer half-life than iodine (Cs-137 has a half-life of 30.0 y) and is another major contributor to personnel and equipment–property hazards. The less-volatile species may be divided broadly into three groups: the semivolatiles (tellurium and antimony), the low volatiles (strontium, barium, and europium), and the refractories (molybdenum, ruthenium, zirconium, cerium, neptunium, etc.).

Approximately 35 elements and over 200 different isotopes are formed in the fission process. Many of these isotopes have sufficiently short half-lives and therefore do not figure predominantly in the amount of

[a]For work performed at the Massachusetts Institute of Technology.

radioactivity present approximately 1 day after shutdown. What complicates time-dependent source term release analyses (especially for the case of Chornobyl's 10-day active phase) is that the longer lived fission products continue to decay until a stable product is formed. The physical and chemical states of the intermediate species in a given decay chain are important because their volatilities span the entire range noted previously. There can be little question that the chemical form of the fission products has a profound effect on their release from fuel during reactor accidents and on their subsequent behavior.

The behavior of accident-released fission products is further complicated (and quite dependent) on the exact nature of the accident—the releases are by no means immediate. Mechanisms such as diffusion through the lattice of the ceramic fuel, mass transfer through a boundary layer-type concentration gradient near a phase boundary in a molten fuel, and gas-phase diffusion external to the fuel have definite time-dependent transport rates (which are strong functions of ambient conditions) that lend themselves to calculations, provided a reasonably descriptive model can be formulated [for example, fission products (which may include fine particulate matter) are transported by liquids, gases, or two-phase mixtures encountered along the release path]. The relative transport by such fluids and possible deposition on interior surfaces is quite dependent on the physical and chemical forms of the release product and surface in question. Most fission products, however, have varying chemical characteristics that affect both adsorption and desorption rates on surfaces, so analytical descriptions of transport and deposition become quite complicated.

The two physical properties most relevant to a source term release analysis are the vapor pressure of the compounds that can form under accident conditions and the free energies of formation of these compounds. The latter indicates the stability of the compounds at elevated temperatures and are of considerable importance in the prediction of the form of released fission products. (This is true not only because UO_2 fuels are used in most nuclear power reactors but also because oxygen is likely to be present in the environment of accident-ruptured fuel materials.) These properties are strong functions of temperature and are significantly affected by the atmosphere and by time. (Burnup, irradiation temperature, and grain size of the fuel are also known to affect releases but to lesser extents.) For example, the most volatile fission products—krypton, iodine, and cesium—are released almost totally at the highest temperatures with little effect of atmosphere; but the releases of fission products such as strontium, molybdenum, ruthenium, tellurium, antimony, barium, and europium are quite sensitive to ambient conditions.

Oxidation of the fuel may be a fission-product release mechanism in severe accidents if the pressure vessel or containment is breached, as was the case at Chornobyl. This mechanism becomes significant if fission products escape from finely divided fuel droplets that are formed as the result of a steam explosion. The release is due to extensive oxidation of the droplets upon their dispersal into an air atmosphere. Additionally, burning of either a metallic fuel or a lower oxide greatly enhances fission-product release by increasing the exposed surface area many orders of magnitude as well as by local overheating and gas expulsion. Any solid material, when heated to a temperature sufficient to induce surface oxidation, will disperse a small quantity of fine particles to the atmosphere. At temperatures approaching the melting, ignition, or boiling point of the material, the process is accelerated.

Thermodynamic data apply, strictly speaking, only to equilibrium conditions that seldom, if ever, exist in reactor accidents. Consequently, it is necessary to be cautious in the use of such data to predict the behavior of fission products under accident conditions. It appears probable, however, that, at elevated temperatures resulting from loss-of-coolant accidents (LOCAs) in reactors fueled with high melting point materials, equilibrium will be at least approached so that conclusions based on thermodynamic considerations are of some value.

It is not within the scope of this article to rigorously model the release mechanisms summarized previously. Rather, the article will attempt to shed some light on Chornobyl radioisotope release dynamics by clearing up inconsistencies in previously published (Soviet) release data and comparing them with new release data presented here. By analyzing and combining new data with information presented in the previous two articles,[1] release curves over the period of the active phase are produced. Finally, a scenario is presented for what may have transpired during this 10-day release period.

CHORNOBYL SOURCE TERM RELEASE ANALYSIS

Appraisal of Soviet Release Data

In their report to the International Atomic Energy Agency (IAEA) in August 1986,[2] the Soviets provided a table of relative releases for various radioisotopes as a function of time during the active phase of the accident (reproduced in Table 1). Unfortunately, there are a number of problems that make some of the data unusable, or, at the very least, of questionable reliability.[3] First, there are several missing data points, which, especially in the case of the volatile and biologically important Cs-137, make it difficult to analyze release rates over the early part of the active phase. Second, there are two large gaps in the data between April 26 and April 29 (2 days missing) and April 29 and May 2 (2 days missing) that further complicate the analysis. Third, the release data for Te-132 and I-132 are combined, which makes it impossible to determine even relative activities of these isotopes; hence they cannot be used in a release analysis over the active phase. Fourth, the method of collection is not fully known, which makes an assessment of the reliability of the data nearly impossible [for example, the data will obviously be sensitive to the precise times, number of, and positions (elevation and ground coordinates) of the sample collections, what filters were used if any, etc.]. Fifth, the data were not properly normalized to 100% on any of the days, but especially for April 29 and May 5.[4] If the data were properly normalized, it would have been possible to estimate values for some of the missing data points, but one may not simply normalize the existing data for a given day if that day has missing data.[5] Sixth, efforts on the ground to contain the graphite fire and limit the consequences of the accident (dropping material on the destroyed reactor and tunnel excavation underneath the building) may have artificially increased the amount of contamination in the air above the reactor. The Soviets themselves mention this possibility (Ref. 2, p. 18).

A seventh reason for doubting the reliability of the data requires a separate explanation and concerns the data corresponding to April 26 (column one, or the first day). There is a question as to whether the data for this column are valid at all because it is difficult to imagine that air samples would have been taken above the reactor so soon after the accident.[6] This is especially true given the unpreparedness of the Soviets for the scale of the accident, the lack of basic dosimetric equipment at the station, and the fact that the next column of data represents samples taken 2.5 days after the accident.[7] Moreover, the director of the Chornobyl Station at the

Table 1 Fractional Activity (∂_i) of 17 Radioisotopes in Air Samples Taken Over Unit 4 (Ref. 2)

	Activity[a]					
Radioisotope	**April 26 (0.5 d, 12 h)**	**April 29 (3.5 d, 84 h)**	**May 2 (6.5 d, 156 h)**	**May 3 (7.5 d, 180 h)**	**May 4 (8.5 d, 204 h)**	**May 5 (9.5 d, 228 h)**
Zr-95	0.044	0.063	0.093	0.006	0.070	0.200
Nb-95	0.006	0.008	0.090	0.013	0.082	0.180
Mo-99	0.037	0.026	0.020	0.044	0.028	0.037
Ru-103	0.021	0.030	0.041	0.072	0.069	0.140
Ru-106	0.008	0.012	0.011	0.031	0.013	0.096
I-131	0.056	0.064	0.057	0.250	0.082	0.190
Te-132 + I-132	0.400	0.310	0.170	0.450	0.150	0.086
Cs-134	0.004	0.006	0.006	0.016	0.006	
Cs-136	0.003	0.004	0.005	0.009		
Cs-137			0.014	0.037	0.013	0.022
Ba-140	0.032	0.041	0.080	0.033	0.130	0.120
La-140	0.110	0.047	0.150	0.023	0.190	0.170
Ce-141	0.014	0.019	0.076	0.009	0.064	0.150
Ce-144	0.016	0.024	0.061		0.051	0.110
Nd-147	0.014	0.017	0.025		0.021	0.054
Np-239	0.230	0.030	0.110	0.006	0.028	0.068
Total	0.995	0.701	1.009	0.999	0.997	1.623
ΣA_i (Ci/L)	3.60×10^{-7}	3.20×10^{-7}	5.00×10^{-8}	7.00×10^{-8}	1.00×10^{-6}	7.00×10^{-9}

[a] $\partial_i = A_i / \sum_i A_i$.

time of the accident, Viktor Bryukhanov, in the early hours of the morning reported that the reactor was still intact—a myth that persisted for many hours and caused not only delays in the evacuation of the plant and the surrounding areas but also downplayed the need for specialists to arrive quickly to monitor the situation.[8] As a consequence, the Soviet government's Chornobyl Commission neither "officially" nor accurately ascertained the extent of the accident until at least after 3:00 p.m. the day of the accident.[9] In all probability (given the circumstances), properly collected air samples above the reactor were not taken on the first day. In fact, a number of the scientists who analyzed data coming in during the first few months after the accident confirm this.[10] The data supposedly representing air samples taken 12 or so hours after the accident actually represent very roughly averaged and integrated soil samples collected later and "fit" to a standard model.

Presentation and Appraisal of New Release Data

Fortunately, new data have been made available from the PROBA data bank system at the Kurchatov Institute of the Russian Scientific Center.[11] These new data are shown in Table 2 and represent samples collected by a Soviet military helicopter flying at an altitude of 200 m over the reactor shaft (building coordinates L-47) with a zig-zag pattern. Unfortunately, from all that can be gathered, the sample collection method employed here and managed by a unit of the Soviet Army's special Chemical Warfare Division may not have been properly carried out.[12] A sample collection system dubbed "Gondola" used three cigar-shaped plastic canisters approximately 1 m long and 20 to 30 cm in diameter (with one opening) attached to the undercarriage of a Soviet AN-26 helicopter. It is not clear what types of filters were used, although it is almost certain that charcoal-activated ones were *not* used—which made the filter effectively "invisible" to gaseous releases. (Apparently, proper filters were used only after the active phase.) Considering the gaseous and chemical behavior of these elements, and even if the data were corrected to account for the low efficiency of the filters for trapping gaseous species,[13] a significant amount of error would have been introduced. Of the filters used, those from the Ministry of Defense were sent to Semipalatinsk (a military installation) for analysis, whereas those from the State Committee of Hydrometeorology (*Goskomgidromet*) were sent to Obninsk near Moscow. Some of the latter filters were analyzed at the Kurchatov Institute in Moscow, and the results of (ostensibly) all analyses were combined to form the data shown in Table 2. (Although it is not clear what time of day these air samples were taken, it may be assumed for purposes of the analysis that the samples were taken at 12 o'clock in the afternoon.)

The PROBA data set does, however, have several advantages over the data presented in Vienna. First, there is an additional day of data (April 28) that Table 1 does not contain. Second, data for Te-132 and I-132 are separated. Third, the data (given as absolute activity values) are well normalized and present release data for more radionuclides over a longer period. (Data are given through May 23 and, although not usable for an active phase analysis, provide releases for almost a month after the accident.) Interestingly, there are no data for April 26, which supports the conjecture that first-day data presented by the Soviets in Vienna were not derived from air samples but rather from integrated ground depositions. Moreover, it is puzzling that both data sets look remarkably similar concerning missing data points, which leads one to speculate that the Vienna conference data are a combination of the PROBA data and integrated estimates for the first day.[a]

Unfortunately, the PROBA data also share some of the same unknowns or problems associated with the data in Table 1 as well as present new difficulties [for example, the fact that fires generally do not burn uniformly and therefore release smoke in puffs or small "bursts" (density variations) makes the data appear quite meager given that it was gathered only once per day at one location—if indeed that is the case]. (Hot particles undoubtedly had substantially different transport dynamics as compared with gases and elemental releases—made even more significant if the samples collected were not taken in the release plume.) On the basis of the weather conditions at the site during the accident and the high heat content of the plume, McInall estimates the transport elevation to correspond to the 850 mbar pressure level, or approximately 1500 m, whereas the maximum mixing level, where

[a]One of my colleagues at Chornobyl is sure this is the case. He claims data were *purposely* "denormalized" strongly on certain days while data for day 2.5 were eliminated altogether to hide the fact that the Ministry of Defense gathered the data. This is, of course, speculation, but it fits the general pattern of Soviets withholding or manipulating information and again raises the question of the reliability of the data and the trustworthiness of Soviet accounts of the accident.

Table 2 PROBA Fractional Activity (∂_i) Release Data From Air Samples Taken 200 m Over Unit 4

Radioisotope	Activity																					
	April 28 (2.5 d, 60 h)	**April 29 (3.5 d, 84 h)**	**May 2 (6.5 d, 156 h)**	**May 3 (7.5 d, 180 h)**	**May 4 (8.5 d, 204 h)**	**May 5 (9.5 d, 228 h)**	**May 6 (10.5 d, 252 h)**	**May 7 (11.5 d, 276 h)**	**May 8 (12.5 d, 300 h)**	**May 9 (13.5 d, 324 h)**	**May 11 (15.5 d, 372 h)**	**May 13 (17.5 d, 420 h)**	**May 14 (18.5 d, 444 h)**	**May 15 (19.5 d, 468 h)**	**May 16 (20.5 d, 492 h)**	**May 17 (21.5 d, 516 h)**	**May 18 (22.5 d, 540 h)**	**May 19 (23.5 d, 564 h)**	**May 20 (24.5 d, 588 h)**	**May 21 (25.5 d, 612 h)**	**May 22 (26.5 d, 636 h)**	**May 23 (27.5 d, 660 h)**
Sr-89		0.0899			0.0263				0.0004													
Sr-90		0.0056			0.0032																	
Zr-95	0.0581	0.2629	0.0927	0.0062	0.0677	0.1243	0.1273	0.0553	0.0276	0.0820	0.1518	0.0198	0.1261	0.0510	0.1943	0.2129	0.1385	0.0805		0.1649	0.0249	
Nb-95	0.0681	0.0000	0.0897	0.0129	0.0795	0.1095	0.1115	0.0457	0.0121	0.1048	0.1802	0.3932	0.1416	0.0322	0.2105	0.2352	0.1611	0.1180		0.1806	0.0302	
Mo-99	0.0422	0.0083	0.0200	0.0443	0.0270	0.0167	0.0104	0.0582	0.0497	0.0150		0.0075		0.0245	0.0020		0.0119	0.0279			0.0252	
Ru-103	0.0397	0.0765	0.0411	0.0772	0.0667	0.0895	0.0728	0.2236	0.2302	0.1159	0.0898	0.3161	0.0476	0.2112	0.0961	0.0735	0.1254	0.2825	0.4141	0.1538	0.4234	0.2487
Ru-106	0.0115	0.0206	0.0110	0.0315	0.0133	0.0592	0.0428	0.0671	0.0633	0.0392	0.0796	0.1005	0.0331	0.1040	0.0344		0.0606	0.1165	0.3233	0.0561	0.1788	0.6490
Te-132	0.1436	0.0570	0.0709	0.1904	0.0561	0.0532	0.0331	0.0611	0.0709	0.0626	0.0194	0.0083		0.0178	0.0017		0.0055			0.0104		
I-131	0.1295	0.0714	0.0564	0.2494	0.0792	0.0706	0.0463	0.1345	0.1797	0.1755	0.0773	0.0651	0.0300	0.2921	0.0153		0.1033	0.2430	0.2626	0.0939	0.2847	0.1023
I-132	0.0649	0.0673	0.0940	0.2561	0.0922	0.0516	0.0228	0.1099	0.1050	0.0756	0.0322	0.0097		0.0322			0.0135	0.0303		0.0169	0.0328	
Cs-134		0.0107	0.0060	0.0162	0.0053		0.0081	0.0096	0.0094	0.0079		0.0023										
Cs-136	0.0034	0.0038	0.0056	0.0090				0.0053	0.0061													
Cs-137			0.0140	0.0370	0.0128	0.0136	0.0114	0.0216	0.0242	0.0195		0.0051		0.0245	0.0071		0.0111					
Ba-140	0.0573	0.0760	0.0797	0.0319	0.1229	0.0736	0.0587	0.0305	0.0432	0.0364			0.1250		0.0554							
La-140	0.1103	0.1756	0.1491	0.0232	0.1893	0.1021	0.1391	0.0507	0.0918	0.0719	0.1501	0.0121	0.1958	0.1389	0.1313	0.1277	0.1385			0.0945		
Ce-141	0.0454	0.0320	0.0760	0.0087	0.0621	0.0910	0.1200	0.0611	0.0331	0.0614	0.0762	0.0227	0.1539	0.0716	0.1176	0.1656	0.0911	0.0540		0.1037		
Ce-144	0.0384	0.0358	0.0607		0.0491	0.0699	0.1260	0.0611	0.0357	0.0715	0.1433	0.0256	0.1470		0.1323	0.1851	0.1137	0.0473		0.1252		
Nd-147	0.0281	0.0065	0.0245		0.0200	0.0334	0.0228	0.0047	0.0102	0.0226		0.0120			0.0021		0.0257					
Np-239	0.1593		0.1089	0.0058	0.0275	0.0418	0.0470		0.0072	0.0382												
Total	0.9998	0.9999	1.0003	0.9998	1.0002	1.0000	1.0001	1.0000	0.9998	1.0000	0.9999	1.0000	1.0001	1.0000	1.0001	1.0000	0.9999	1.0000	1.0000	1.0000	1.0000	1.0000
Σ Activity (Bq/m^3)	369 820	194 729	48 960	74 570	1 009 200	10 777	1 524	51 877	85 144	32 428	17 589	10 249	18 643	404	380 936	7 186	21 292	6 337	1 234	46 020	2 142	792

vertical dilution of the plume reaches ambient conditions, was at approximately 3000 m.[14] (NUREG-1250, however, reported that, beginning on approximately April 28, the elevation of the plume did not exceed 200 to 400 m.[15]) Third, air sample filter efficiency plays an important role in the "capture" of the various physical forms the releases may have taken. If, indeed, "improper" filters were used to collect samples during the active phase, they would have collected hot particles and aerosols depleted in volatile elements; at the same time, they would have been quite transparent to gaseous species—in effect "missing" them. Devell et al. indirectly support the significance of filter efficiency for Chornobyl by reporting that "60% to 90% of iodine captured in their samples [in Sweden] was in a gaseous form or a form desorbable from particles rather than as particulate cesium iodide."[16] Fourth, the PROBA data are *assumed* to be uncorrected for radioactive decay (that is, presented are the radioisotopes' absolute release activities). Even this is not positively confirmed, however. Finally, fifth, the PROBA data do not agree with similar data shown in Table 4.7 of the Soviet report—in some cases, release activities are off by an order of magnitude (Ref. 2, p. 8). This appears to further call into question other data presented by the Soviets at Vienna.

Difficulties notwithstanding, the PROBA and Soviet/IAEA data sets may be combined by limiting ourselves to the active phase and assuming that the PROBA data are the more reliable of the two (that is, data from 12 hours is added to the PROBA set from the Soviet Vienna report because it does not record sample data for this period). Also, because PROBA contains only two data points for Sr-90 over the active phase, we will not consider them in the combined set. Table 3 contains the combined normalized data sets, whereas Table 4[17] contains volatility characteristics of these isotopes and compounds that may have formed during the period of the active phase.

Active-Phase Release Dynamics

Radioisotope release is generally considered a classic vaporization process: the driving force for vaporization is the disequilibrium between the fuel–debris mixture and the ambient gas. If this is truly the case, releases driven by the vapor pressures of the radioisotopes are expected to display substantial release rate differences. If not, it suggests that some factor other than vapor pressure is driving (or limiting, as the case may be) the rates of radioisotope release. By converting the activity release data in Table 3 to releases in terms of mass, a statistical (correlation) analysis shows that the mass releases correlate quite well

Table 3 Combined Data for the Fission-Product Fractional Activity (∂_i)

	Activity							
Radioisotope	**April 26 (0.5 d, 12 h)**	**April 28 (2.5 d, 60 h)**	**April 29 (3.5 d, 84 h)**	**May 2 (6.5 d, 156 h)**	**May 3 (7.5 d, 180 h)**	**May 4 (8.5 d, 204 h)**	**May 5 (9.5 d, 228 h)**	**May 6 (10.5 d, 252 h)**
Zr-95	0.044	0.058	0.291	0.093	0.006	0.070	0.124	0.127
Nb-95	0.006	0.068		0.090	0.013	0.082	0.109	0.111
Mo-99	0.037	0.042	0.009	0.020	0.044	0.028	0.017	0.010
Ru-103	0.021	0.040	0.085	0.041	0.077	0.069	0.090	0.073
Ru-106	0.008	0.012	0.023	0.011	0.032	0.014	0.059	0.043
Te-132	??	0.130	0.079	0.056	0.249	0.082	0.071	0.033
I-131	0.056	0.144	0.063	0.071	0.190	0.058	0.053	0.047
I-132	??	0.065	0.074	0.094	0.256	0.095	0.052	0.023
Cs-134	0.004		0.012	0.006	0.016	0.005		0.008
Cs-136	0.003	0.003	0.004	0.006	0.009			
Cs-137				0.014	0.037	0.013	0.014	0.011
Ba-140	0.032	0.057	0.084	0.080	0.032	0.127	0.074	0.059
La-140	0.110	0.110	0.194	0.149	0.023	0.195	0.102	0.139
Ce-141	0.014	0.045	0.035	0.076	0.009	0.064	0.091	0.120
Ce-144	0.016	0.038	0.040	0.061		0.051	0.070	0.126
Nd-147	0.014	0.028	0.007	0.025		0.021	0.033	0.023
Np-239	0.230	0.159		0.109	0.006	0.028	0.042	0.047
Total activity, $A_{i,k}$ (Bq/m^3)	13 320 000	369 820	176 139	48 960	74 570	979 460	10 777	1 526

Table 4 Basic Radiochemical Information for Radioisotopes with PROBA Release Data[17,a]

Isotope	Volatility classification	Core inventory at t = 0, Ci × 10^6	Half-life	GMM	Elemental M.P., °C	Elemental B.P., °C	Oxide compound	Oxide M.P., °C	Oxide B.P., °C
I-131	V: halogen	83.2	8.04 d	130.9061	113.5	184.3	IO_2, IO_4, (CsI)	d 75–130 (626)	*b*, (1280)
I-132	V: halogen	121	2.284 h	131.9080	113.5	184.3	IO_2, IO_4, (CsI)	d 75–130 (626)	*b*, (1280)
Te-132	V: telluride	121	78.03 h	131.9085	449.57	988	TeO, TeO_2	d 370, 733	d 1245
Cs-134	V: alkali metal	4.60	2.062 y	133.9068	28.39	671	Cs_2O, Cs_2O_2	>400, 400	*b*, 650 -O_2
Cs-136	V: alkali metal	3.10	13.16 d	135.9073	28.39	671	Cs_2O, Cs_2O_2	>400, 400	*b*, 650 -O_2
Cs-137	V: alkali metal	7.01	30.0 y	136.9068	28.39	671	Cs_2O, Cs_2O_2	>400, 400	*b*, 650 -O_2
Ru-103	RM: noble metal	102	39.25 d	102.9063	2334	4150	RuO_3, RuO_4	d 25.5	*b*, d 108
Ru-106	RM: noble metal	23.2	372.56 d	105.9073	2334	4150	RuO_3, RuO_4	d 25.5	*b*, d 108
Mo-99	RM: noble metal	165	65.76 h	98.9077	2623	4639	MoO_2, Mo_2O_5	6.47, *b*	*b*, *b*
Zr-95	RO	159	64.02 d	94.9080	1855	4409	ZrO_2	*ca* 2700	*ca* 5000
Nb-95	RO	153	34.97 d	94.9068	2469	4744	NbO_2, Nb_2O_5	*b*, 1520	*b*, *b*
Ba-140	MV: alkaline earth	164	12.746 d	139.9106	729	1805	BaO, BaO_2	1918, 450	*ca* 2000, 800 -O_2
La-140	NV: rare earth	164	40.16 h	139.9094	918	3464	La_2O_3	2307	4200
Ce-141	RO: rare earth	150	32.5 d	140.9082	798	3443	Ce_2O_3, CeO_2	1692, 2600	*b*, *b*
Ce-144	RO: rare earth	105	284.9 d	143.9136	798	3443	Ce_2O_3, CeO_2	1692, 2600	*b*, *b*
Nd-147	RO: transuranic	58.5	10.98 d	146.9161	1021	3074	Nd_2O_3	≈1900	*b*
Np-239	RO: transuranic	1570	2.355 d	239.0529	639	3902	NpO_2, Np_3O_3	*b*, d 500	*b*, *b*

[a]V, volatile, MV, moderately volatile; NV, nonvolatile; RM, refractory metal; RO, refractory oxide; GMM, gram molecular mass; M.P., melting point; B.P., boiling point; d, decomposes; and *ca*, around.

[b]Data are unavailable or unknown.

for all 17 isotopes over the active phase. This may be depicted graphically by plotting the ratio of the released (sampled) masses to the calculated masses in the fuel (i.e., normalizing, assuming no release) shown in Figs. 1 and 2. The isotopes are separated broadly into two groups (volatiles and nonvolatiles) on the basis of an intriguing divergence in the behavior of the two groups during days 6.5 to 7.5. Additionally, for ease in comparison, the "bathtub" curve is reproduced as Fig. 3.

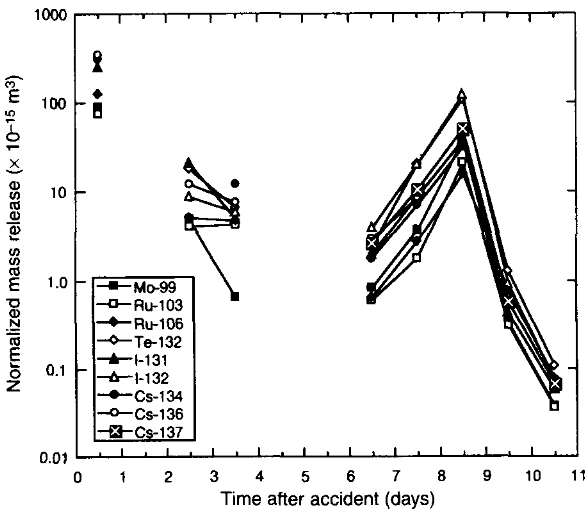

Fig. 1 Normalized mass release over the period of the active phase for the volatiles.

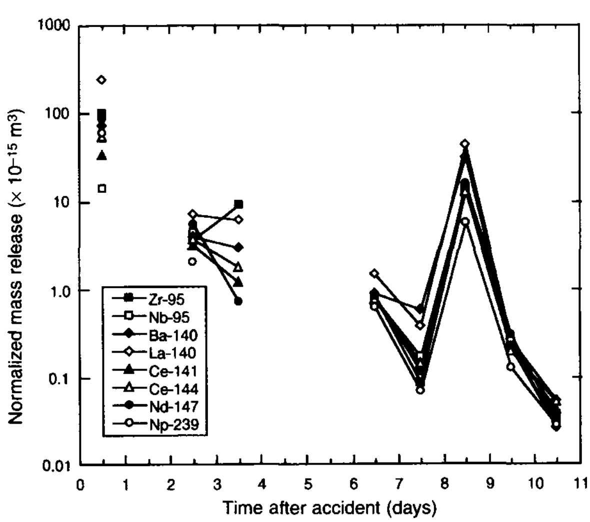

Fig. 2 Normalized mass release over the period of the active phase for the nonvolatiles.

The first thing to note is how closely the release behaviors of all 17 isotopes follow each other: with the exception of the peak on day 8.5, all decrease over time. Generally, the volatiles are releasing about one-half an order of magnitude higher than the nonvolatiles. This seems to suggest that the releases were occurring isothermally over the period of the active phase. Of course, as the decay heat of the corium decreased and heat energy continued to escape, one would have *expected* that, for an *uncovered* core, the releases would decrease over time. (Significant temperature variations would lead to very large differences in the release behaviors—at least between the volatiles and the nonvolatiles.) Interestingly, ruthenium and molybdenum, usually considered non- and mid-volatiles, respectively, behave much like the volatiles. (Mo-99 has a substantial and mysterious drop in its releases on day 3.5—which may be due to sample collection or measurement error.)

The second thing to note is that, with the exception of day 7.5, the releases are (somewhat unexpectedly) all within one and one-half orders of magnitude of each other—which suggests that during this period some mechanism was either substantially limiting the release of the volatile radionuclides, substantially enhancing the release of the nonvolatiles, or some combination of both.

A third interesting feature of these curves is that the peak in releases near the end of the active phase occurs at 8.5 days after the accident. This seems to disagree slightly with the peak of releases as depicted in the "bathtub" curve. From Figs. 1 and 2 (because of the 1-day frequency of data collection), however, it is clear that an even higher peak in releases could have occurred any time after day 8.5 and before day 9.5. Although, as noted previously, the reliability of the first day's data is somewhat questionable,[a] nevertheless, the general shape of these curves corresponds quite well to the "bathtub" curve with the (assumed) initial peak caused by steam-explosion ejection and another peak near the end. The peak in releases around day 8.5 is followed by a sudden and

[a]This is not to imply, however, that releases were low on the first day. Given the relative kinetics of the oxidation of zirconium and graphite, a very high first-day(s) release could be postulated to be associated, at least partially, with a very rapid zirconium burning followed by the somewhat slower burning-off of the graphite. This hypothesis supports the general shape of the mass release curves, at least during the first few days when the graphite may have served to retain some releases. It is also supported by the "red and blue fire" and "powerful updraft" coming from the mouth of the reactor crater as reported by witnesses [see Ref. 7(b), p. 103].

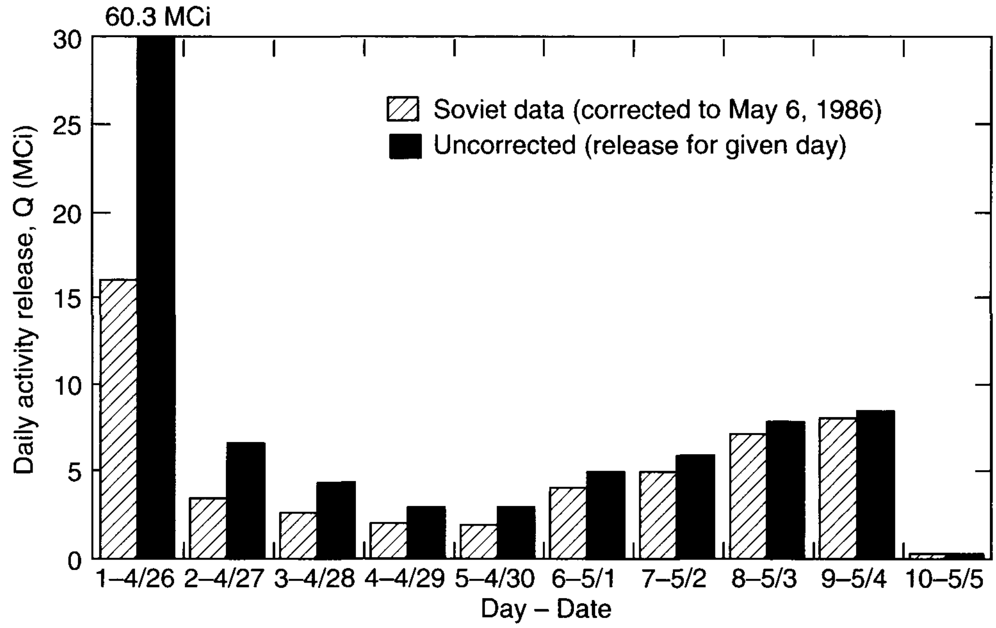

Fig. 3 Release rate curve presented by the Soviets in August 1986 at Vienna: the "bathtub" curve.

significant drop around day 9, which indicates the termination of releases and the end of the active phase. All the release behaviors of all the isotopes converge rapidly after day 8.5.

Finally, the most dramatic difference between the two sets of curves occurs during days 6.5 to 7.5: without exception, all the volatile isotopes display a significant increase in releases, whereas the nonvolatiles show a marked decrease. (Despite missing data on day 7.5 for Ce-144 and Nd-147, because both are refractory oxides and Ce-144's sister isotope, Ce-141, which behaves as the other nonvolatiles, we can be fairly certain that cerium and neodymium may be grouped into the nonvolatiles for this analysis.) This divergence in behaviors is followed by a strong convergence to form the peak on day 8.5. Subsequently the isotopes display almost identical behaviors following this peak.

The most plausible explanation for the divergence in behaviors (taking into account information presented in previous articles) is that the graphite may have burned off at approximately day 7. This would enhance the release of the volatiles because (bearing in mind the temperatures expected in the corium mass) the filtering effect of an (assumed) upper graphite layer would be gone. It would also decrease the releases of the nonvolatiles because there would be a weaker particulate release mechanism available as a pathway to the environment.[18] The peak at day 8.5 could be explained by the possibility that at that time the lower biological shield (LBS) may finally have been melted through—followed by a rapid relocation of approximately 135 tonnes of corium to the lower regions of the reactor building. The relocation (physical mechanism) and spread (larger surface area) of the corium would have provided greater opportunity for releases. The larger surface area would also, however, provide for more rapid cooling. This, together with the fact that the chemistry of the corium would have been complicated by taking up the approximately one-quarter section of the LBS (raising its solidus) and the fact that significantly less decay heat was available as input energy by day 8.5, would make the corium much less able to interact with surrounding materials and give it a propensity for rapid solidification.

Samples of hot particles analyzed both within the former Soviet Union and in the West show that, except for volatile radioisotopes, the nonvolatiles were released more or less in fuel form (that is, the isotopic content and character of material released are heavily skewed toward the nonvolatile radioisotopes and actinides, which implies that these radioisotopes did not vaporize but were transported to the environment within the matrix of small fuel particles). Conversely, to explain why some particles were enriched in cesium (Ref. 3, p. 14), a certain fraction of the released particles may have served as "carriers" onto which the volatiles condensed. As mixtures of vapors (volatiles) and mechanically produced aerosols (containing nonvolatiles in the matrix) cool, the surfaces of the aerosols are preferred sites for condensation of the volatiles because of the high surface-area-to-volume ratio (approximately 3000 cm^{-1}).

The total amount of radioactive "dust" or hot particles contained within the sarcophagus is estimated to be 5 ± 2 tonnes.[1(b)] It is likely that this dust formed when hot, molten fuel (melted as a result of the power excursion) accelerated from the rupture of fuel rods into the cooling water in the pressure tubes, disintegrated into aerosol-sized droplets as the result of hydrodynamic instabilities. Another possible mechanism for the comminution of fuel is its expulsion into an oxidizing environment. During the oxidation of UO_2 to U_3O_8, there is a change in the crystal structure that ruptures the fuel grains—if the fuel is not already melted. The exposed surfaces, therefore, provide a direct path for the vaporization of volatile radioisotopes that is further enhanced by virtue of a high surface-area-to-volume ratio of the particle Also, the pulverized fuel particle itself is available for transport out of the fuel mass, restricted by bulk gas flow rate over the surface of the fuel and filtering material (if any) between the fuel and the environment. The conclusion then is that, depending on ambient conditions, the surface of the fuel or hot particle can enhance either vaporization or condensation of volatile radioisotopes—which implies that the composition of aerosols is not a reliable indicator of the mechanical aerosolization process. Therefore Soviet release rate data as presented in Vienna (Ref. 3, p. 16) along with Western analyses of hot particle fallout from Chornobyl should be viewed with caution inasmuch as they reflect two release mechanisms strongly influencing releases during the active phase of the accident—the vaporization of volatiles and aerosol transport of hot particles.

The Soviets have typically presented their release analyses in terms of a dimensionless "fractionation factor," k_{i-j}, defined in the following manner: in the event of a severe nuclear reactor accident in which there is melting or severe destruction of the core and in which the containment or reactor building is breached, a certain fraction of the fission products will escape the fuel (vaporize) whereas another fraction will remain trapped within the fuel or within a fuel-and-structural material admixture. The activity of any fission product remaining in fuel-containing masses (FCMs) may be related to the total calculated activity for that fission product as

$$A(t)_i^{\mathrm{FCM}} = \xi(t)_i A(t)_i^{\mathrm{TOT}} \tag{1}$$

$$A(t)_i^{\mathrm{REL}} = [1 - \xi(t)_i] A(t)_i^{\mathrm{TOT}} \tag{2}$$

where $A(t)_i^{\mathrm{TOT}}$ = total calculated core inventory of nuclide "i"

$A(t)_i^{\mathrm{FCM}}$ = nuclide "i" activity remaining in FCM

$A(t)_i^{\mathrm{REL}}$ = nuclide "i" activity released from FCM

$\xi(t)_i$ = fuel binding coefficient, $0 \le x_i \le 1$ (i.e., for full release $\xi_i = 0$)

Of course, all these are complex functions of time whose physical basis depends on the half-life of the nuclide, the chemical and thermodynamic properties of the element, and its interactions with surrounding materials. Consequently, for any nuclides "i" and "j," the fractionation factor (k_{i-j}) may be defined with the use of Eqs. 1 and 2 as

$$\frac{A_i^{\mathrm{REL}}}{A_j^{\mathrm{REL}}} = \frac{(1-\xi_i)}{(1-\xi_j)} \frac{A_i^{\mathrm{TOT}}}{A_j^{\mathrm{TOT}}} \tag{3}$$

or

$$\frac{(1-\xi_i)}{(1-\xi_j)} = \frac{(A_i/A_j)_{\mathrm{REL}}}{(A_i/A_j)_{\mathrm{TOT}}} = k_{i-j} \tag{4}$$

where $0 \leqslant k_{i-j} \leqslant (1/1 - \xi_j)$ if ξ_j is known.

This provides a practical tool for the dimensionless analysis of radioisotope releases. If the initial core inventory can be determined just prior to the accident (i.e., A_i^{TOT}, where "i" is any isotope), if further the amount of a certain long-lived nonvolatile "tracer" fission product in the melted fuel can be measured or "fixed" (i.e., A_j^{FCM}), and, finally, if the isotopic composition of releases relative to the total activity released (i.e., $A_j^{\mathrm{REL}}/\Sigma A$) can be measured, it is possible to characterize the efficacy or volatility of release of a particular radioisotope over time with respect to the tracer isotope with the use of the fractionation factor. Note, however,

Table 5 Radioisotope Fractionation Factor ($k_{i\text{–}95}$) During the Active Phase

		Activity							
Radioisotope[a]		April 26 (0.5 d, 12 h)	April 28 (2.5 d, 60 h)	April 29 (3.5 d, 84 h)	May 2 (6.5 d, 156 h)	May 3 (7.5 d, 180 h)	May 4 (8.5 d, 204 h)	May 5 (9.5 d, 228 h)	May 6 (10.5 d, 252 h)
Zr-95	RO	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Nb-95	RO	0.14	1.17		0.93	1.98	1.10	0.82	0.81
Mo-99	NM	0.91	1.27	0.07	1.00	41.9	2.97	1.28	1.00
Ru-103	RM	0.74	1.08	0.46	0.72	20.3	1.62	1.19	0.95
Ru-106	RM	1.23	1.32	0.52	0.76	32.3	1.23	2.97	2.07
Te-132	V:T		4.81	0.72	2.95	239	8.52	5.06	2.87
I-131	V:H	2.51	5.66	0.54	2.37	103	2.99	1.66	1.53
I-132	V:H		2.34	0.66	4.77	239	9.62	3.59	1.91
Cs-134	V:A	3.11		1.35	2.09	83.5	2.48		1.91
Cs-136	V:A	3.45	3.20	0.83	3.54	98.7			
Cs-137	V:A				3.16	124	3.88	2.23	1.76
Ba-140	AE	0.72	1.06	0.32	1.09	6.83	2.51	0.85	0.70
La-140	RE	2.40	1.87	0.68	1.80	4.36	3.39	1.04	1.44
Ce-141	RO	0.34	0.84	0.13	0.92	1.60	1.05	0.85	1.10
Ce-144	RO	0.54	0.97	0.29	0.93		1.01	0.78	1.36
Nd-147	RE	0.88	1.49	0.08	1.00		1.24	1.19	0.84
Np-239	RO	0.61	0.56		0.75	0.80	0.46	0.50	0.73

[a]RO, refractory oxide; NM, noble metal; RM, refractory metal; V:T, volatile:tellurium group; V:A, volatile:alkali; V:H, volatile:halogen; AE, alkaline earth; and RE, rare earth.

that this tells little about whether once released from the fuel, the fission products may become trapped by debris (plate out on a cooler surface) or chemically interact with structural or other reactor and building materials before escaping into the environment where they are detected.

Several tables of fractionation factors over the period of the active phase have been published by the Soviets.[a] Unfortunately, either the data are incomplete or, in some cases, the values for the same tracer isotope have been inconsistent between studies, which makes the results of their analyses inconclusive.[b,19] The tracer isotope chosen for the present analysis is Zr-95 because it has a relatively long half-life compared with the period of the active phase (64.02 d decaying to Nb-95) and because its release activities are all present in the combined data set (Table 3). Table 5 contains the fractionation factors calculated for the PROBA data. In Figs. 4 to 6, these data are graphically divided into three groups: the volatiles, the refractory metals, and the refractory oxides.

Certain trends are apparent from the fractionation factor curves. First, as expected, the less volatile a species, the lower its $k_{i\text{–}95}$ value (i.e., the lower its efficacy for release with respect to Zr-95). The difference in magnitudes of releases between the three groups is significantly less than expected, however. In fact, the differences between the volatiles and refractory oxides (which both roughly bound the refractory metals) are within an order of magnitude of each other. This suggests that some mechanism is indeed limiting the release of volatiles during the active phase—which, in turn, is supported not only by the normalized mass release curves (Figs. 2 and 3) but also by previous findings that an average 35% of the Cs-137 inventory in the corium was retained.[1]

Although it is difficult to determine conclusively (because there are no data available for the period between days 3.5 and 6.5 for comparison), it is clear that some peculiarity during April 28 to 29 (days 2.5 to 3.5) sharply decreased the release efficacy of all the radioisotopes. Given the way air sampling was conducted (essentially at one position in space and once per day), this may simply be a result of a wind shift at the

[a]The first such method for presenting release data, using Zr-95 and Ce-141 as the tracer isotopes, was published in the proceedings of the IAEA/Soviet conference at Vienna in 1986: Ref. 2, Appendix 4, Table 4-11, p. 14.

[b]See Ref. 2.

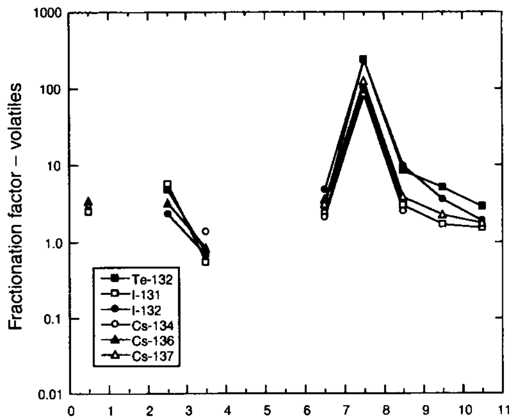

Fig. 4 Fractionation factor (k_{i-95}) for volatiles over the active phase.

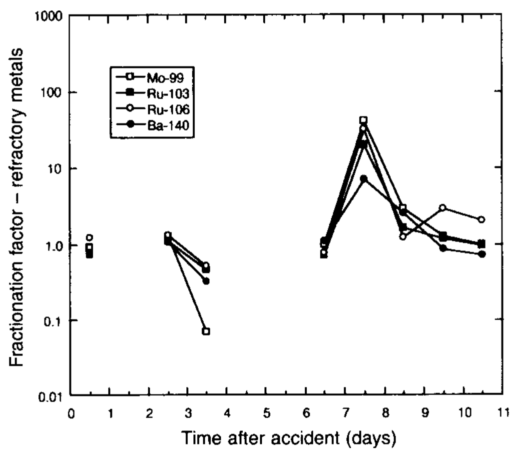

Fig. 5 Fractionation factor (k_{i-95}) for refractory metals over the active phase.

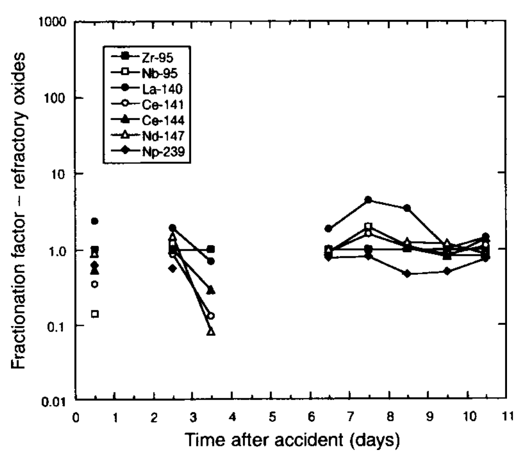

Fig. 6 Fractionation factor (k_{i-95}) for refractory oxides over the active phase.

time of the sampling.[a] Another possibility may be that a very small fraction of the material being dumped by the helicopters was actually able to enter the core shaft and disrupt the geometry enough to inhibit (for some reason) the release efficacy with respect to Zr-95.[b] The effects of these possible scenarios cannot now be determined. Without more information we may only speculate.

A more probable explanation for the sharp drop in fractionation factors during days 2.5 to 3.5 is as follows. The normalized mass release curves (Figs. 1 and 2) indicate a decrease for all radioisotopes except for the tracer Zr-95, which *increased* substantially from day 2.5 to 3.5. Additionally (although bearing in mind the reliability of data for the first day is somewhat questionable), Nb-95 in Fig. 6 shows an almost order of magnitude increase in its fractionation factor. Because the core of a 1000-MW(e) RBMK reactor contains approximately 103 tonnes of Zircaloy-2.5 (Zr-2.5% Nb) in the channel pressure tubes and 74 tonnes of Zircaloy-1 (Zr-1.0% Nb) in the fuel rods[20] (yielding 173.5 tonnes of zirconium metal and 3.5 tonnes of niobium metal in the core), the relatively large release may be caused by the excess presence of these metals. At the time of the accident, the fuel contained 5.8 and 3.9 kg of Zr-95 and Nb-95, respectively. Moreover, neutron activation could have produced significant amounts of these isotopes in the fuel cladding and pressure tubes (relative to fission products produced in the fuel) to explain such high k_{i-95} values.[21] If this indeed is the explanation, one would then expect a smoother transition in the fractionation curves during this period and by extension a flatter normalized mass release rate.

As with the normalized mass release curves, the most prominent feature of the fractionation factor curves is the peak near the end of the active phase where the volatiles show the greatest increase in efficacy of release while the refractory oxides show the

[a]In fact, it was on these days that a major shift in the wind occurred. During the first 3 days, the prevailing winds were blowing toward the north-by-northeast, carrying radioisotopes into Scandinavia. Afterward, the wind shifted to the southeast, carrying the release plume over the Ukrainian capital Kyiv.

[b]Recall that the helicopter dumping of materials during the active phase occurred from April 27 to May 2 (days 1.5 to 6.5) and that trace amounts of lead were found in the lava-like fuel containing materials (corium) (see Ref. 1, Table 4, in Sich and Table 4 in Borovoi and Sich).

least. In contrast to the former curves, however, the latter peak one day earlier at day 7.5. Although it is known that the dumping of materials into the reactor building stopped on May 2 (day 6.5), it is hardly likely that this could have influenced the releases given the fact that the material did not cover the major part of the core (≈135 tonnes) still located in the reactor shaft *at that time*. The difference in peak positions is more readily explained by the fact that, according to the normalized mass release curves, Zr-95 has the second lowest value on day 7.5. Dividing by such low normalized value of activity to obtain the fractionation factors would artificially inflate $k_{i\text{–}95}$ for the isotopes, especially in the cases of the volatiles. This is confirmed for day 8.5, where the fractionation factors return approximately to their day 6.5 values.

Because the fractionation factor is known, little can be said about the absolute release rates for the radioisotopes in question because $k_{i\text{–}95}$ represents only the "efficacy" of isotope release relative to the Zr-95 tracer. Note also that, in this analysis, the methodology differs from a similar analysis in Powers et al. (Ref. 3, pp. 14-15), where the data from Table 4.10 combined with Table 4.13 of the Soviet report[a,2] (the "bathtub" curve) were used to produce figures of fractional release rates for individual radioisotopes. [It has already been shown that the Soviet accounts of accident management actions and release data presented at Vienna are inaccurate.[1(a)]] The results for the analysis presented were obtained by combining newly available PROBA data with calculations that provided the core inventory in intervals of 12 hours for 300 hours after the accident. Results from the normalized mass release curves show that radioisotope release rates remain more or less steady or even decrease with time—in contrast to Powers et al.,[3] where release rates are shown to increase significantly.

Source Term Release Estimate

A nonrigorous estimate for volatile radionuclide and total releases from Chornobyl may be made by combining the results of radiochemical analyses of fuel-containing masses (FCMs or "lava") and material balance applied to the remaining fuel located within the sarcophagus[1] together with results of the release analysis presented here. Analyses of the corium located in the lower regions of the reactor building show that the ratio of the measured activity of Cs-137 to the calculated (or "expected") activity (A_{meas}/A_{calc}) is 0.35 ± 0.11. Unfortunately, although the previous article in this series presents a fuel material balance, the errors are large, and apparently some of the fuel is unaccounted for. The corium located in the lower regions of the reactor building has been studied and quantified more extensively than other forms of the fuel. Even here, however, efforts to determine the amount of fuel present have not been very precise because the extreme radiation environment within the sarcophagus has forced researchers to employ rather ingenious methods (without independent confirmation) to locate and quantify fuel in inaccessible regions of the reactor building.[22] Moreover, there has been no positive confirmation (although it is fairly certain) that approximately 11 tonnes of nuclear fuel is located on the floor of the Central Hall of the reactor building beneath 5020 tonnes of materials thrown from helicopters in an attempt to smother the burning core. Even if eventually there is no fuel found there, the "missing mass" falls well within the bounds of uncertainty.

The volatile radionuclides are the halides (bromine and iodine) and alkali metals (rubidium and cesium) along with the metals silver, tellurium,[b] and antimony. The estimate, however, is limited to iodine, cesium (most biologically hazardous), and tellurium because release data are available. Out of these isotopes, further restriction may be made to those with significant half-lives (that is, those with half-lives on the order of 1 d or greater).[23] Because approximately 65% of the cesium was released from the lava, an estimate must also be made of the amount of iodine[c] and tellurium released—and all three are subject to the actual amount released beyond the reactor building (that is, a certain amount of the releases plated out onto debris surfaces in the damaged building). If it is assumed (in fact, underestimated) that only 35% of the cesium from the lava was released to the atmosphere (i.e., approximately half

[a]The hypotheses concerning what may have been happening in the reactor core within the reactor shaft during the active phase are, of course, flawed. See Ref. 2, Part II, Appendix 4, pp. 13 and 20.

[b]Note the volatility of tellurium will be depressed somewhat if it combines with metallic zirconium.

[c]To confirm how much iodine was released from this portion of the fuel, one should analyze the corium for the presence of I-129 as suggested in the previous article. However, it would be difficult to detect I-129 because of its long half-life (1.574×10^7 years) and because it emits a weak beta (0.15 MeV) and gamma (39.6 keV) that would be lost in the "noise" of other fission-product decays in the corium.

plated out),[24] it may also be assumed that 50% × 1/2 plate out radiotellurium and 80% × 1/2 plate out radioiodine were released.[a] Finally, consideration must be given to the approximately 14% (27 tonnes) of fuel that is likely located on the floor of the Central Hall as well as to other fuel scattered about as a result of the explosion. Because the helicopters did not start dropping materials into the Central Hall until around 10:00 a.m. on Sunday, April 27 (at least 32 hours after the accident), and because visual evidence suggests that this portion was "burning" (the main reason for attempting to smother it with the materials), it is expected to have contributed significantly to overall releases. The assumption is that releases from this portion of the core were 30% Cs, 35% Te, and 50% I [i.e., they were roughly proportional to the ratios reported by the Soviets in Vienna (that is, 13% Cs, 15% Te, and 20% I)].[b]

Table 6 contains release estimates for the eight most significant volatile isotopes on the basis of the available information. Most significant is that the release estimate for these *eight isotopes alone* is approximately 92 MCi—which is substantially more than a *total* release of 50 MCi (excluding noble gases) claimed by the Soviets in Vienna in August 1986. If one then considers the fact that releases were very great during the first day (as a result of the nature of the explosive forces), that the plate-out fraction estimated in Table 6 is more than likely too high, and if the contributions of all other longer lived radioisotopes are added, the total release may approach 150 MCi.[25] In fact, if Np-239 (half-life 2.355 d) is considered and if it was released at the 3.2% fraction claimed by the Soviets,[2] its contribution to the releases over the period of the active phase *alone* could reach 30 MCi.

Of course, it is not sufficient (nor proper) to characterize the source term release simply by the magnitude of radioactivity releases to the environment. For an estimation of the release, among other things, the half-lives of the released radioisotopes as well as their relative biological toxicities must be considered. Although the activities of the various fission products and transuranics formed in the core are great, many of these isotopes have short half-lives and may be of no real consequence. Other fission products, most notably those of tellurium, decay rapidly, but in doing so transmute to other isotopes of equal or greater hazard (namely iodine). Again, special note should be made of Np-239 because so much of it is produced that it contributes 30% or more of the total gamma activity in the core at shutdown.

One final point must be made concerning the Soviet claim that 3.5 ± 0.5% of the core *mass* (6660 ± 950 kg) was released beyond the bounds of the station. With the use of the initial core inventories of the isotopes contained in Table 6, it is readily shown that the initial mass of these isotopes (560 MCi activity) is about 100 kg. The implication is that, even if all these isotopes are considered and if all are 100% released from the core, this mass is lost in the "noise" of the 950-kg error in the Soviet estimate, which further implies that considerable *activity* releases may have occurred that would be virtually undetectable in the *mass* release estimate. By emphasizing the 3.5 ± 0.5% release as correct (which admittedly was correct in terms of mass), attention may have been diverted away from biologically significant radioisotope releases.

RECONSTRUCTION OF EVENTS DURING THE ACTIVE PHASE

Summary of Previous Results

Rather than simply an academic exercise, establishing the sequence of events for the active phase of the Chornobyl accident has important ramifications for western severe reactor accident and degraded core analyses as well as for source term release analyses. Moreover, on the basis of the new data and information presented in this and the previous two articles, a radically different account appears to be emerging as to what may have happened to the Chornobyl Unit 4 core during the active phase.

The first article in this series revisited and reappraised Accident Management Actions (AMAs) taken to contain the release of radioisotopes into the

[a]The ratio of releases from the fuel of these three elements (Cs/Te/I = 65/50/80) reflects roughly the release ratio as presented by the Soviets in Vienna (Cs/Te/I = 13/15/20). This is not a bad estimate and, in fact, may be low given the volatility of iodine and conditions for release ("dry" ambient conditions and no filtration). As volatiles are carried away from the bulk corium in an atmosphere of steam and hydrogen, some may condense on structural surfaces or on aerosols formed from vapors of structural materials. However, given the high ambient heating—especially in the region of the reactor shaft—little could have plated out on hot structural debris.

[b]Note that one may not simply neglect as insignificant volatile radionuclide releases from the large core fragments and certainly not from microparticles created as a result of the accident. Recent studies have shown that this may not be the case, and releases may have been quite significant (relative to the bulk mass release).

Table 6 Estimated Volatile Isotope Lower-Bound Activity Releases from Chornobyl [Radionuclides with Significant Half-Lives ($t_{1/2} \geq 1$ d)][a,b]

Basic/initial (t = 0) data				Fuel-containing materials (FCMs)[c]				Central Hall and outside reactor building[d]				Total
Isotope	Half-life	Activity, MCi	Mass, kg	Fractional contribution[e]	Fractional release[f]	Fractional (1-plate out)[g]	Release,[h] MCi	Fractional contribution	Fractional release	Fractional (1-plate out)	Release, MCi	Total release, MCi
Te-129m	33.6 d	28.1	0.93	~0.71	(0.50)	(0.5)	5.0	~0.29	(0.35)	(0.9)	2.6	7.6
Te-132	78.03 h	121	0.40	~0.71	(0.50)	(0.5)	12.0	~0.29	(0.35)	(0.9)	6.2	18.2
I-129	1.57×10^7 y	(2.0×10^{-6})	(11.3)	~0.71	(0.80)	(0.5)	Neg.	~0.29	(0.50)	(0.9)	Neg.	Neg.
I-131	8.04 d	83.2	0.67	~0.71	(0.80)	(0.5)	16.8	~0.29	(0.50)	(0.9)	7.7	24.5
I-133	20.8 h	146	0.13	~0.71	(0.80)	(0.5)	25.7	~0.29	(0.50)	(0.9)	11.8	37.5
Cs-134	2.062 y	4.6	3.5	~0.71	0.65	(0.5)	1.1	~0.29	(0.30)	(0.9)	0.4	1.5
Cs-136	13.16 d	3.1	2.3	~0.71	0.65	(0.5)	0.4	~0.29	(0.30)	(0.9)	0.1	0.5
Cs-137	30.0 y	7.0	80.4	~0.71	0.65	(0.5)	1.6	~0.29	(0.30)	(0.9)	0.5	2.1
Total		559	99.6				62.6				28.0	91.9

[a]Figures in parentheses are estimates—note that the fractional release estimates are probably quite low and the plate out estimates are probably high.
[b]The release estimates for Te-132, I-133, I-131, and Cs-136 were modified (reduced) to take into account radioactive decay over the active phase.
[c]FCMs are that portion of the core currently located in the lower regions of the reactor building.
[d]"Central Hall and outside reactor building" refers to that portion of the core located on the floor of the Central Hall, outside the reactor building but within the bounds of the station, and beyond the bounds of the station.
[e]Mass fraction of the entire core contributing to this release.
[f]Fractional release from the particular portion of the fuel in question (as obtained or estimated from radiochemical analyses).
[g]Fraction that escaped from the fuel and debris into the environment.
[h](Initial activity corrected for decay over the active phase) * (fractional contribution) * (fractional release) * [fractional (1-plate out)].

environment during the active phase. In particular, it is clear that Soviet attempts to smother the fire by dumping materials onto the Central Hall failed to cover the destroyed core. The implication is that the core burned virtually in the open—which implies further that substantially more radioactivity was released than reported by the Soviets at Vienna. Additionally, given that the other AMAs outlined in the first article were also, for the most part, unsuccessful in containing that portion of the core participating in the corium–lower biological shield (LBS) melt-through,[a] the Chornobyl accident may well define the upper bound for severe accident releases.

That more radioactivity was released is supported by results of radiochemical analyses of corium presented in the second article of the series. Approximately 71% of the core (~135 tonnes) melted through the LBS, flowed into the lower regions of the reactor building, and quickly solidified into several forms of ceramic glass and pumice-like substances. The amount of Cs-137 remaining within the corium matrix as a fraction of the initial inventory is approximately 0.35 [that is, approximately 65% of the initial inventory *of this portion of the core* (~71%) may have been released]. There are two curious features concerning this result. First, it clashes with the 13 ± 6.5% Cs-137 release reported by the Soviets in 1986 at Vienna (Ref. 2, Part II, Appendix 4, p. 21). Second, that fully 35% of the initial inventory of Cs-137 was retained in the fuel is unexpected given the length of time this portion of the core may have been molten (~9 days) while exposed to strongly oxidizing conditions, expectations of Cs-137 retention based on elemental volatility, and the experience of TMI-2 where only between 3 and 19% of the inventory of Cs-137 was retained in the molten debris under *reducing* conditions.[1] Unfortunately, it is not known in what chemical form the cesium was found. If it formed the nonvolatile CsI (which is soluble in water), this may explain the relatively large fraction retained in the corium at Chornobyl and why so little was retained at TMI-2. This, however, is unlikely or at least not significant because there was not enough iodine in the fuel to form CsI—even more unlikely given that a large portion of the iodine was released to the environment.

Finally, the fact that the LBS acted as a "core-retainer," permitting fuel decay heat generation to decrease while complicating the chemistry of the corium–LBS admixture, effectively reduced the ability of the corium to interact with surrounding structural materials. The first and second articles in this series

[a]Recall that the approximately 135 tonnes of UO_2 fuel participating in the melt-through is equivalent to the full core load of a 1300-MW(e) western boiling-water reactor (BWR).

detailed the state of the nuclear fuel located in the lower regions of the reactor building and showed quite clearly, albeit unexpectedly, that, after having melted through the LBS, the corium did relatively little damage even to surrounding metallic structures. This result appears to justify western efforts to design core-retention components as part of future light-water reactor containment designs.[a] It may also eventually show that the infamous and quite exaggerated "China Syndrome" appears much less likely to occur, even for an accident as severe as Chornobyl.

By combining the evidence presented in all three articles of the series, it is now possible to hypothesize what may have occurred in the Chornobyl Unit 4 core during the period of the active phase.

Mechanical–Dynamic Stage

The starting moment for this stage (01:23:39–40 on the morning of the accident) is actually 4 to 5 seconds prior to the beginning of the active phase and is defined as the time at which the operator pressed the emergency scram (AZ-5) button, an action for which to this day the motivation has not been clearly established. Unknown to the operators, the effect of the reinsertion of this fully withdrawn control-rod bank (AR and RR rods) was to add approximately $+0.5\beta$ to an already out-of-control reactivity increase initiated by the coastdown of the main coolant pumps participating in the safety experiment. It was about this time that the Central Hall shift foreman [who was located approximately 14.5 m (Level 50 m) above the floor of the Central Hall] witnessed a significant event: not only did he feel the strong and frequent shocks (as did the reactor operators located on Level 10 m) but also he noticed that the 2488 fuel and control channel caps (each with a mass of 80 kg located directly over the reactor at floor level in the Central Hall) were vibrating violently up and down [see Ref. 7(b), p. 74, and Ref. 26].

Apparently, by this time more than one or two pressure tubes in the reactor had bursted, which released steam into the reactor space and overpressurized the thin steel reactor pressure boundary, Component KZh.[b] Steam would then have entered the space above the upper biological shield (UBS) but below the floor-level shield blocks covering the pressure-tube refueling connections. If the steam blowdown were rapid, some shield blocks could even have been ejected off the floor. Moreover, a breach of Component KZh would have permitted steam to easily lift the 2500-tonne-equivalent UBS and its associated coolant piping.[c] Even a slight rise (1 to 2 cm) of the UBS would have severely strained (if not sheared) most of the channels at the weakest points in the coolant circulation circuit—the zirconium-to-stainless steel transition welds located above and below the core. It is speculated here that the first "explosion" may actually have been the rise and fall of the very large and heavy UBS and the associated rapid blowdown of steam as it escaped from the reactor cavity.[d]

By this time the fuel channels would have been sufficiently damaged as to inhibit or even block the further insertion of the emergency scram rods into the core. Additionally, with the shearing of the channels, there would have been a rapid flashing to steam of coolant in the reactor and a severe overpressurization of the reactor space and possibly adjacent compartments. It is conjectured here that the extremely rapid overpressurization of the reactor space was subsequently followed by its explosive release, hurling the UBS approximately 10 m into the air and rotating its lower face partially upward. Judging from the current state of the UBS and the belief that a significant amount of fuel (on the order

[a]Note that a typical 1000-MW(e) western pressurized-water reactor (PWR) or BWR has a UO_2 fuel load of approximately 75 and 115 tonnes, respectively, as compared with the RBMK-1000 with a 190.3-tonne UO_2 fuel load where, for the case of the Chornobyl accident, about 135 tonnes flowed into the lower regions of the reactor building.

[b]"...vault rupture disks are designed to accommodate the rupture of only one pressure tube in [the] reactor... a LOCA to several tubes would be [considered] a severe accident" (Ref. 26, pp. 3-32). Any subsequent local power increase (due to voiding) in the affected channel(s) could propagate to other channels in the immediate area—although not to the degree of the affected channel—exacerbating the problem. Soviet designers consider rupture of a pressure tube *inside the reactor vault [i.e., inside* the graphite pile] to be beyond the design basis of the plant (based on "leak before break"), although a (*Component KZh*) rupture disc is based on one tube rupturing. "The rupture of more than one pressure tube is beyond the design basis of the RBMK-1000 reactor. Such an event would exceed the stated relief capacity of the reactor vault and could overpressurize it. Excess pressure might deform or rupture the vault, or it might lift the Upper Biological Shield enough to relieve pressure to the upper core exit piping region." (Ref. 26, pp. 3-52.)

[c]Reference 26 states, "If the pressure exceeds 0.3 MPa (44 psi or about three atmospheres), the Upper Biological Shield will lift up. Since the fuel channels are welded to the upper shield, its upward movement will lead to massive tube failures. Furthermore, since the control rod channels are also connected to the UBS, the control rods will be lifted out of the core." It is clear that little excess pressure is required to induce a catastrophic failure of the channels.

[d]An everyday example of this is the cover of a violently boiling cooking pot "dancing" as steam built up on the inside overcomes the force (weight) of the cover and escapes to ambient.

of 11 tonnes[1]) may be located 15 m to the east-southeast of the reactor cavity under the 5000-tonne pile of debris thrown from the helicopters attempting to "smother" the core, the UBS probably carried along with it a small portion of the core still connected by pressure tubes that were not severed during the initial lifting of the UBS. This portion of the core was then ejected into the Central Hall; it was disconnected from the UBS in midflight and landed approximately over the cover panels of the southern spent-fuel cooling pond.

That a portion of the core may have landed in the Central Hall in a particular direction of core fragment ejection appears to be supported by three pieces of evidence. First, as viewed from above the damaged reactor, immediately after the accident core graphite moderator and reflector blocks along with their associated fuel lay scattered on the roof of Unit 3 and the auxiliary building in a "preferred" direction to the east-by-southeast. In fact, before these roofs were decontaminated, a distinct "anti-shadow" lay behind the large ventilation chimney located on the roof of the auxiliary building (Block "V") where the chimney had blocked core debris from landing behind it in an east-by-southeast direction while all around lay graphite blocks. Second, as described previously,[1] it is now clear that the infamous "red glow"—initially thought to be the burning core located inside the reactor cavity—was presumably only a small portion of the core located above the southern spent fuel pool.[a] Finally, damage to the reactor building as evidenced by structural members and heavy equipment displaced from their normal positions appears to indicate that a particularly strong explosive pressure wave was directed toward the east-southeast, possibly partially reflected off the lower face of the UBS as it was in midflight. Interestingly, that the UBS was in the air for a certain period of time is evidenced by the facts that it is currently resting on a damaged portion of the high bay wall at the entrance to the reactor cavity in the southwest quadrant[1] and that other portions of panels from the high bay wall are now located at the bottom of the reactor shaft on top of the remains of the LBS.

Particularly intriguing is how these high bay wall panels ended up at the bottom of the reactor shaft, especially considering that the explosive release of steam (that hurled the UBS upward and completely destroyed the upper parts of the Central Hall) was directed outward as evidenced by the remains of the reactor building.[27] It is possible that a hydrogen–steam mixture entered the reinforced steam-drum separator chambers (Rooms 804/3 and 804/4—by way of ducts along which the coolant piping exiting the top of the reactor eventually reaches the separators) and attained detonation concentrations on the order of 4% hydrogen. This would explain how the wall panels would have been blown in toward the middle of the Central Hall.[b]

[a]An intriguing hypothesis states that the "red glow" was neither burning core graphite, burning bitumen from the roof, nor a hot, glowing portion of the core. Rather, it may simply have been the *image* of the exposed and very hot core still remaining in the reactor shaft *reflected* off the tilted UBS and onto debris located to the east of the reactor shaft. As helicopter crews attempted to bomb this "glow," the pile of materials would have been increasing in size, thus presenting a larger and larger area upon which the image of the core could be reflected. Additionally, this would explain why the "glow" didn't disappear shortly after being bombed from the air and would also explain why some witnesses stated they saw the "glow" even after the bombing campaigns were stopped on May 2—7 days after the accident. This hypothesis appears to be credible for two reasons: (1) the operators that entered the Central Hall to appraise the damage shortly after the accident did not mention any burning or glowing mass located away from the reactor shaft and (2) the thick "smoke" emanating from the Central Hall during the period of the active phase would have obscured visibility for the helicopter crews.

[b]Grigori Medvedev assertively conjectures that this "series of explosions destroyed the drum-separator compartments, as well as the drum separators themselves... tearing them from their attachments and from the pipelines." In fact, although some of the wall panels from the separator chambers facing inward toward the Central Hall are missing (apparently blown out), the steam drums themselves are *not* torn off their supports nor away from the coolant lines. Apparently, Medvedev assumed that whatever hydrogen was produced in the core traveled along the coolant channel piping into the steam drums themselves. Besides the fact that the steam drums are not heavily damaged, it is difficult to fully agree with this hypothesis because the channels in which hydrogen was being produced would have been destroyed, making it difficult for hydrogen transport along these channels. Moreover, it is easier to imagine that, rather than along the coolant channels, hydrogen would have traveled along the ducts containing these channels [see Ref. 7(b), pp. 81-82].

Medvedev is convinced that the major explosion was the detonation of hydrogen and that this is what destroyed the reactor building and hurled the UBS into the air: the detonation occurred in the reactor,"which was full of hydrogen." There are at least two problems with this hypothesis. First, it is unclear if enough hydrogen was formed in the core in such a short period to have produced this magnitude of explosion. Recall from the first article in this series that apparently two narrow power surges occurred at 01:23:44 and 01:23:46. This is also the time at which the explosions were heard. Before the initial power surge, the power was rising relatively slowly. It is therefore difficult to imagine that much hydrogen could have formed before 01:23:44. Second, even if a great deal of hydrogen was formed during and after the initial power surge, the detonation would have required the presence of oxygen in the core—also difficult to imagine.

Finally, as concerns the possibility for hydrogen detonation in the steam drum chambers, some time would have been required for any hydrogen produced in the core (together with oxygen from the same source) to have entered the chambers and to have attained detonation concentrations. This again leaves open the question of how the high bay wall panels managed to enter the reactor shaft: if hydrogen explosions occurred in the steam drum chambers, they conceivably must have occurred *after* the major steam explosion that hurled the UBS into the air. If this is the case, how could the panels have entered the reactor shaft unless the UBS was in the air for a very long time?

Radioactivity Release Stage

Setting the Stage. The analysis now turns to the behavior of the core *after* the initial releases that resulted from the explosive nature of the destruction of the core. It is clear that these short-term initial releases are extremely difficult to model directly. By necessity they must be estimated from ground deposition measurements together with a thorough modeling of active-phase release rates. It is beyond the scope of the present analysis to provide a rigorous model of these release rates. An effort was made earlier, however, to clarify the behavior of certain radioisotopes over the course of the active phase; it is hoped that future studies will be able to more accurately estimate these release rates. The intent here is to provide a reasonable account (hypothesis) of what may have transpired within the core shaft during the active phase. There are several important points (conclusions drawn from the preceding information) upon which the hypothesis is based:

1. Little or none of the materials thrown by helicopters during the first 6 days after the accident in an attempt to "smother" the burning core made it to that major portion of the core (≈71%) located within the reactor shaft.

2. Results of radiochemical analyses indeed seem to confirm that essentially no lead or boron carbide made it into the core shaft. In addition, visual evidence (photographs) indicates that the materials thrown from the helicopters formed a pile located approximately 15 m to the east of the core shaft and that there is little or no evidence of these materials covering the opening to the reactor shaft or the UBS.

3. Nitrogen purging of the core region was apparently started after the active phase. This therefore proved to be ineffective except as a backup in the event that the hot corium continued its downward movement.

4. A detailed description of the forms, compositions, and locations of the corium (LFCMs) permitted visualization of the dynamic behavior of the LFCMs.

5. New Soviet release data and an account of how data were collected permitted not only a reappraisal of the Soviet data but also a better analysis of the behavior of 17 isotopes whose chemical characteristics varied greatly. This, in turn, provided a rough notion of what may have been occurring to that portion of the core located within the reactor shaft over the period of the active phase.

The Hypothesis. The preceding results significantly alter conceptions of what may have been occurring in the core region during the active phase of the accident and form the basis for the hypothesis. The hypothesis is presented as a sequence of events defined by a series of six stages (Fig. 7).

Stage 1: The initial period after the explosive destruction of the core is Stage 1. The portion of the core that did not blow up and out of the core region eventually settled to the bottom of the core shaft on top of the LBS. The UO_2 and core metal structures (lower steel supports and Zircaloy tubing) also settled to the bottom and eventually melted. The vigorous oxidation (exothermic) of the graphite and zirconium (as evidenced by witnesses' accounts of a "howling" rush of air emanating from the mouth of the reactor cavity) together with decay heat generation (an integrated energy release estimate yields 8.5 GJ over a 10-day period) would have greatly suppressed the plate out of volatilized radionuclides or those carried by particulates. The heating of structures in the region (especially the lower portions of the UBS) would thus allow for significant releases to occur.

Stage 2: From the time that the core and associated core structures settled to the bottom of the reactor shaft to approximately 6.5 to 7 days later, differences in the densities of the components of the pulverized graphite–corium mass would cause a differentiation and layering effect. The melted fuel would have first formed a lower layer below the graphite. There it would interact initially with the stainless steel plating of the LBS and then with the stainless steel coolant piping and serpentine filler (recall, a hydrous magnesium silicate in the form of sand or small pebbles) within the LBS. Eventually, the fuel itself would differentiate to form a lower metal-enriched layer and a middle oxide layer with the graphite forming the top layer only slightly insulating the rest of the corium. That a lower metal-rich layer was formed is evidenced by the current forms and locations of the LFCMs.[a] What little (if any) sand or dolomite did get into the core—together with the serpentine and cast iron–pebble filler material for the LBS—combined with the lava and increased the melting point of the corium mixture, which lowered its ability to interact and slowed the melting process. Heating of surrounding structures as the result of graphite oxidation and decay heat generation would continue to suppress radionuclide plate out.

[a]The second article in this series indicates that a thin lower layer of highly radioactive solidified metal (especially enriched in Ru-103, Co-60, and other metals) exists under a thicker ceramic glass-like layer of corium in the Steam Distribution Corridor (see Ref. 1).

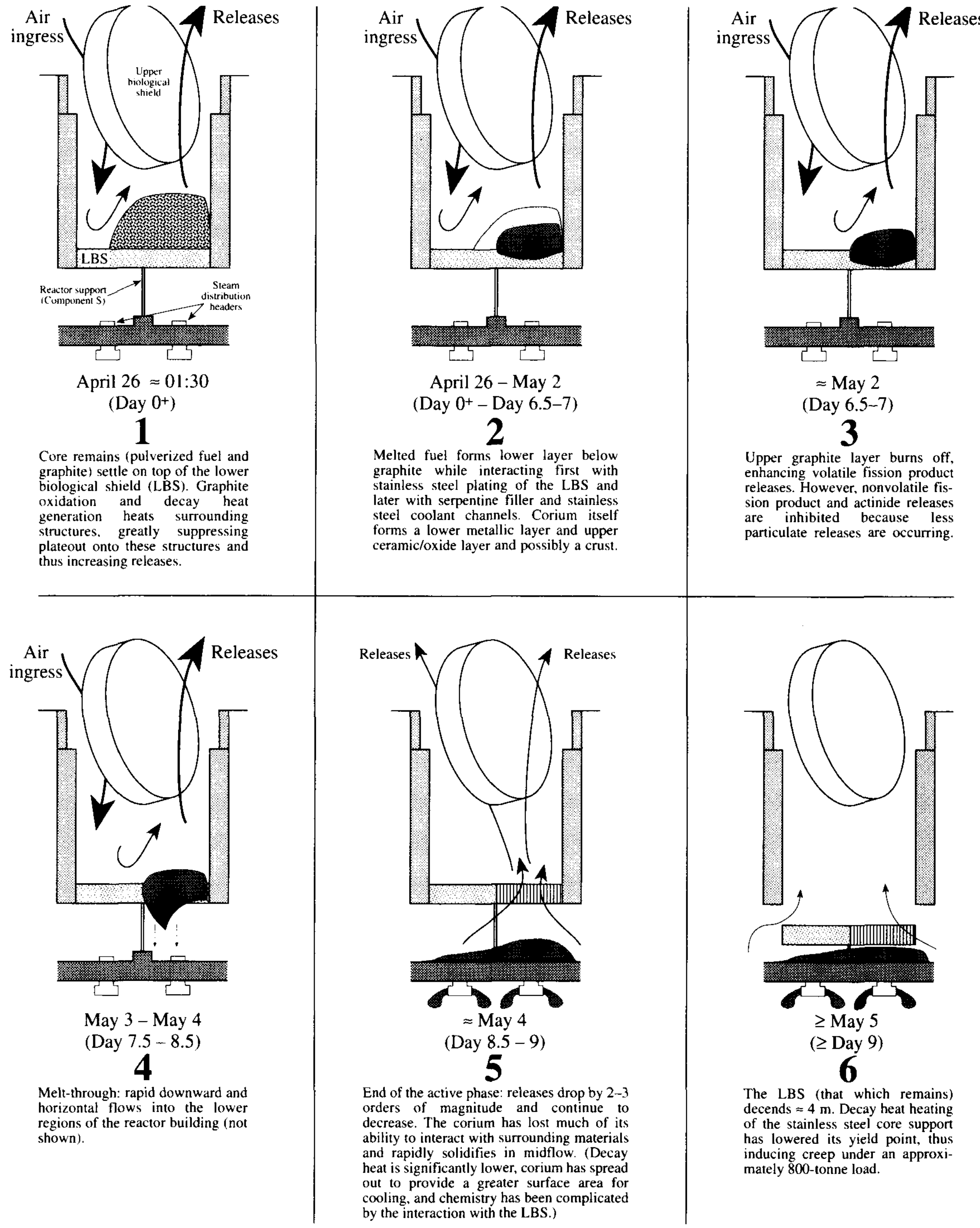

Fig. 7 Hypothesized accident-progression scenario of the period of the active phase (looking north).

Stage 3: After approximately 6.5 days, the graphite associated with 71% of the core lying at the bottom of the reactor shaft (~1340 tonnes) had burned off. There are three reasons to believe this. First, except for a few scattered blocks still located on top of the remaining portion of the LBS, there is no evidence of graphite remaining in the core region except for the charred and blackened southeast wall of the reactor shaft onto which some must have plated out. Second, there is no evidence of carbon or carbon-containing compounds in the results of radiochemical analyses of the corium. Third, there is a distinct deviation in behaviors of the volatile vs. nonvolatile radioisotopes as shown in Figs. 2 and 3, beginning around day 6.5 and ending around day 7.5. Up until that time, the 17 volatile and nonvolatile radionuclides behaved quite similarly, which indicated an isothermal process. It is possible that up to this point the upper layer of burning graphite behaved as a filter that partially retained a significant portion of the volatile fission products being released. Once gone, the release of the volatiles would be enhanced. Conversely, when this graphite "filter" burned away, the nonvolatile radioactive species lost a significant transport mechanism: namely, the transport of nonvolatile radionuclides that may have condensed onto particulate graphite "carriers."

Stage 4: At approximately day 7.5 to 8 days after the accident, the corium melted through the LBS. Careful examination of the LBS shows that roughly its southeast quadrant is completely missing, whereas the remaining edges show signs of melting and sagging. The melt-through was followed by the corium dropping onto the floor of the subreactor region where the force of this rapid redistribution apparently damaged the wall between Room 305/2 and Room 304/3[a] and thus permitted corium to flow into Room 304/3, Corridor 301/5, and eventually southward down Corridor 301/6 [see figures in Ref. 1(b)]. The rapid (mechanical) redistribution of the corium and an increase in its surface area as it spread horizontally substantially enhanced radionuclide releases. This is also evidenced in Figs. 2 and 3, where at day 7.5 there is a marked increase (by almost two orders of magnitude) in the normalized mass releases of the nonvolatiles and an order of magnitude increase in the release of volatiles to day 8.5. The corium also flowed downward into the steam distribution corridor and pressure-suppression pool where it solidified into ceramic glass and pumice-like formations [see figures in Ref. 1(b)]. Still hot, the corium produced steam when it contacted whatever water remained in the pressure-suppression pool and caused an increase in the release of aerosols, which may also partly account for the peak in releases observed on day 8.5.

Stage 5: At approximately 8.5 to 9 days after the accident, the active phase came to an end. Releases dropped by two to three orders of magnitude and continued to decrease. By this time the corium had lost much of its ability to interact with surrounding materials and rapidly solidified in mid flow; this caused little if any damage even to metallic piping in the lower regions of the reactor building. Decay heat had dropped significantly, and the chemistry of the corium had been substantially altered (complicated) by the uptake of one quadrant of the LBS into the molten corium.

Stage 6: Continued cooling of the solidified corium led to further reductions in releases: the hardened surface of the corium and reduced heated air buoyancy effects suppressed releases. At this point (beginning at about day 9), decay heat generation was 50 kW(t)/tonne of uranium, or about 3.75 MW(t) total heat generation. The approximately 75 tonnes of fuel contained in the corium located in Room 305/2 (subreactor region) began to heat the 110-tonne steel reactor support, Component S, which at that point supported an approximately 800-tonne load. The combination of decay heat and loading stresses enhanced creep in the reactor support, which eventually compressed accordion-style; this allowed the LBS to descend about 4 m from its nominal position. The fact that the LBS descended slowly rather than as a result of the initial explosions that destroyed the core is evidenced by the quite smooth, drawn-out condition of the lower coolant channels to the north and south.[27,28]

RECOMMENDATIONS FOR FURTHER STUDIES

Although the hypothesis presented here appears to agree well with the information and data presented, much remains to be studied to verify its plausibility and ultimate acceptance as a valid active-phase scenario. There are five areas of research that should be pursued to confirm its validity. First, the melt-through of the LBS should be accurately modeled to establish a melting rate and time frame for ultimate penetration of the LBS. This should incorporate, in particular, the effect of

[a]Note that the mass of corium has now substantially increased from 135 tonnes of fuel to a complex mixture of metals and serpentine with a mass of approximately 1200 tonnes.

serpentine sparging (which contains more water of crystallization than regular concrete) on corium mixing and fission-product transport mechanisms in the bulk material. Second, rigorous modeling is needed to determine the burn-off rate and ultimate loss of the approximately 1340 tonnes of well-rubblized graphite moderator and reflector that settled to the bottom of the reactor shaft together with the fuel–zirconium mixture.[29] Third, the effect of graphite as a particulate-transfer pathway for volatiles and nonvolatiles should be studied to determine what mechanisms may inhibit the release of radionuclides and to confirm the shape of release curves presented previously. Fourth, studies at Idaho National Engineering Laboratory appear to confirm that more cesium is retained in corium than previously thought [see Ref. 1(b)]. More attention should therefore be focused on understanding high-temperature chemistry effects on cesium and iodine volatility in degraded core analyses and what role oxidizing graphite may have played in apparently suppressing the volatility of cesium. Fifth, because research conducted to date at Chornobyl has provided only a rather rough, descriptive account of the fuel and reactor building, a renewed effort, with the aid of western researchers, should be made to rigorously analyze the remains of Chornobyl Unit 4 as a basis for the preceding recommended studies.

Pazukhin[30] provides an alternate view as to what may have occurred during the active phase. His intriguing study bases its hypothesis on detailing the chemical and thermodynamic properties of the LFCMs found in the lower regions of the Unit 4 reactor building. By conducting a heat balance (heat generation in the fuel and accompanying exothermic chemical reactions vs. heat-transfer losses and endothermic chemical reactions), this study estimates that at approximately 11 hours after the accident, the LFCMs melted through the LBS and began to flow downward into the lower regions of the reactor building (as compared to day 7.5 to 8.5 in this article). Pazukhin concludes that by day 3.5 after the accident the molten LFCMs were cooling rapidly and beginning to solidify and that by day 4.0 to 5.0 the LFCMs were solidified. Given the information presented in this article, it is difficult to fully agree with a number of the conclusions Pazukhin draws in his study, not the least of which is its inability to explain the radioisotope releases over the entire active phase and the sudden drop in releases by three orders of magnitude on days 9 to 10. It is hoped that this series of articles, together with Pazukhin's findings, will encourage further studies and investigations.[31]

Finally, and most importantly, it is clear that the IAEA's International Nuclear Safety Advisory Group (INSAG) should review conclusions drawn in its original INSAG-1 report concerning the AMAs and source term release estimate with the intent of producing a reassessment of the *consequences* of the accident similar to its INSAG-7 follow-up report on the *causes* of the accident. Similarly, conclusions drawn in the IAEA's *International Chornobyl Project* (May 1991) should also be carefully reviewed. This is especially so given that medical experts at a November 1995 World Health Organization conference on the health effects of Chornobyl all but directly linked the marked increase of childhood thyroid cancers and other maladies occurring in Belarus and Ukraine with releases of radioiodine from the accident.

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5. Despite the omissions, Demin and Khodakovsky attempt to "correct" this by normalizing the column of data for April 29th in another publication. (V. F. Demin and I. L. Khodakovsky, Released Radioactive Material During the Chornobyl Accident, as Appendix 10 in *USNRC Workshop on the Chemical Reactions and Processes in Severe Reactor Accidents*, November 1987, Table 12a, p. 151.) Not only is less information presented than in Table 4.10 of the Soviet/IAEA report at Vienna (they neglect to include data for May 4th and May 5th), but in their attempt to normalize without including Cs-137 (one cannot normalize only part of a set of data), they confuse the issue even more. The implication from the normalized data is that no Cs-137 was released on that day! The same normalization is applied to two

of the unnormalized columns of the original Soviet data by D. A. Powers et al. (Table 3, Ref. 3, p. 15), which is subsequently used in their analysis to produce fractional release rate curves.

6. Interestingly, air sampling within the 10-km inner exclusion zone (but not necessarily above the reactor) began on April 29th: "... sent in three groups of specially equipped helicopters, staggered at heights from fifty meters to two kilometers. Every two minutes each helicopter sucked in a sample of the air. By measuring the radioactivity... the active emission every twenty-four hours was about one hundred times higher than that estimated by the physicists on the ground." Piers Paul Read, *Ablaze: The Story of the Heroes and Victims of Chornobyl*, pp. 114-115, Random House, Inc., New York, 1993.
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10. Scientists from Kyiv, Moscow, and St. Petersburg interviewed by Aleksandr Borovoi and Edvard Pazukhin during 1992–1993.
11. For a brief explanation, see Aleksandr Aleksandrovich Borovoi, Fission Product and Transuranic Element Release During [the] Chornobyl Accident (preprint), presented at the International Conference *The Fission of Nuclei—50 Years*, Leningrad, USSR, October 16–20, 1989, Chornobyl, Ukraine, The Chornobyl Complex Expedition of the I. V. Kurchatov Institute of Atomic Energy, pp. 8-9, 1990.
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24. This estimate of the plate out fraction is too high. Directly in the path of the releases from the fuel, the upper biological shield was being vigorously heated from the molten fuel (decay heat, graphite oxidation, and other exothermic chemical reactions). Moreover, recent experimental investigations conclude that "...as a result of their high chemical reactivity, the alkali metals display different transport characteristics in steam and hydrogen. In reducing atmospheres, a major fraction of the cesium is condensable, but *under oxidizing conditions, the aerosol forms predominate* (emphasis added). [Morris F. Osborne and Richard A. Lorenz, ORNL Studies of Fission Product Release Under LWR Severe Accident Conditions, *Nucl. Saf.*, 33(3): 362 (July–September 1992).]
25. The results of a recent study appear to imply that even more may have been released. (L. Devell, S. Güntay, and D. A. Powers, *The Chornobyl Reactor Accident Source Term: Development of a Consensus View*, research publication OECD/GD(96)12, OECD Nuclear Energy Agency, November 1995, pp. 12-14, 25.) If one accounts for radioactive decay over the period of the active phase and adds the contributions of the volatiles I-133, CS-136, and Te-129m as estimated here to that of Devell et al., it can be shown that more than 200 MCi, or four times the initial Soviet estimate, may have been released.
26. The U.S. DOE in NUREG-1250 has commented on this possibility: "The steam released from a main steamline break in either steam separator room would fill the space above the Upper Biological Shield (UBS), and below the shield blocks covering the pressure tube refueling connections. Steam would escape to the refueling floor via the gaps between the shield blocks. The steam would impinge on control rod drive mechanisms and could damage the control rod drive motors sufficiently to prevent operation. If steam blowdown is rapid, some shield blocks could be ejected off the floor." [U.S. Nuclear Regulatory Commission (Comp.), *Report on the Accident at the Chornobyl Nuclear Power Station*, Report NUREG-1250, pp. 3-33, 1987.]
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28. To further support this, a Kurchatov Institute report concludes that it is very unlikely that the total energy available from a steam explosion could have *instantaneously* compressed the reactor support structure by driving the LBS downward. In fact, the authors believe that rather than by a compressive shock, the collapse of Scheme S is more likely to have occurred as the result of a slow (several hours) heating of Component OR, its subsequent warping, and ultimate loss of mechanical integrity. [A. A. Tutnov, A. I. Ulyanov, and A. S. Kiselev, *An Evaluation of the Mechanical Destruction at the 4th Unit of the Chornobyl NPP*, Kurchatov Institute Report 33P/1-517-89, July 28, 1989.]
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General Safety Considerations

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Second ANS Workshop on the Safety of Soviet-Designed Nuclear Power Plants

By R. A. Bari[a]

Abstract: *The* Second American Nuclear Society Workshop on the Safety of Soviet-Designed Nuclear Power Plants *was held in Washington, D.C., in November 1994. The workshop consisted of both plenary sessions and working sessions with 300 participants overall. All countries with operating Soviet-designed nuclear power plants were represented, and representatives from several other countries also participated. In addition to the status and plans related to technical issues, the workshop also included discussions of economic, political, legal, and social issues as they relate to the safety of these nuclear power plants.*

The *Second American Nuclear Society (ANS) Workshop on the Safety of Soviet-Designed Nuclear Power Plants,* which was held in Washington, D.C., in conjunction with the 1994 ANS Winter Meeting, attracted over 300 participants from 25 countries. The first workshop, held 2 years earlier in Chicago, Illinois, in conjunction with the 1992 ANS Winter Meeting, helped to focus the international technical community on the priority issues related to the enhancement of safety of the RBMK and VVER reactors.

Indeed, over the past few years many programs in numerous countries have been aimed at safety improvement of these reactors. For ANS to sponsor the second workshop was therefore timely and appropriate to obtain an international update on safety progress and to identify the important challenges to further safety improvement.

The tragic accident at the Chornobyl Plant about 10 years ago served to focus the world's attention on the safety of Soviet-designed nuclear power plants. Also, much has happened politically and economically to allow for interaction between technologists in the West and in the former Soviet Union and Eastern European countries. Some institutional barriers have been removed, whereas others, resulting from the new ways of doing business, have come to the fore. Liability is the prominent issue.

Throughout this period, however, all parties have demonstrated a genuine desire to ensure that the safety of these reactors be enhanced and maintained at a level such that they could contribute to the world mix of energy sources. The workshop was designed to facilitate the exchange of views and to obtain new information on the safety of Soviet-designed reactors—specifically, the RBMK and VVER designs. It also focused on important issues related to safety improvement.

Safety experts from many countries participated in the working sessions and provided insights, suggestions, experience, and, of course, valuable technical information. Both near-term and long-term programs are needed to ensure that these reactors will be operated safely—and some of these programs were discussed. During this workshop and the Winter Meeting that immediately followed, safety experts and related officials in countries in which RBMKs and VVERs are situated had opportunities to interact, both formally and informally, with a broad cross section of the ANS membership and others from outside the former Soviet

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Union. Learning from the participants from countries with Soviet-designed reactors about the successes in safety enhancement to date and what they regard to be the most pressing challenges was a valuable lesson. Together we must move forward in building an infrastructure that facilitates the safe production from a vital energy source. One important theme of this workshop was safety enhancement to ensure a vital contribution to the world energy mix.

This article provides the summaries of the working sessions and plenary sessions that took place during the workshop (see Table 1). These summaries are based on the written reports of the session rapporteurs.

These written reports, the keynote address by Dr. Terry Lash, and the banquet speech by Dr. Ivan Selin, are contained in the Workshop Summary Report.[1]

KEYNOTE SPEECH BY TERRY R. LASH

In the keynote speech, Terry Lash, Director of the U.S. Department of Energy's Office of Nuclear Energy, noted that nuclear safety is his highest priority. He emphasized the importance of establishing a nuclear safety infrastructure and safety culture in many countries operating Soviet-designed reactors. He also noted that preventing a nuclear accident *in any country* should be a priority for the U.S. nuclear industry because a serious accident would jeopardize opportunities to sell new nuclear power plants both to U.S. utilities and in foreign markets.

He provided a historical perspective by saying that in the spring of 1992 the second of a series of high-level international conferences on coordination of assistance to the former Soviet Union was held in Lisbon, Portugal. At the meeting the United States announced a major financial commitment for improving nuclear safety in Russia and Ukraine. That same year the G-7 leaders met in Munich, Germany, and undertook a serious effort to deal with nuclear safety issues. This program was reaffirmed at Tokyo, Japan, in 1993 and at Naples, Italy, in July 1994. These programs included operational safety improvements, risk-reduction measures, and strengthening of regulatory activities.

In January 1993, the G-7 agreed to create a multilateral fund, referred to as the Nuclear Safety Account, at the European Bank for Reconstruction and Development. This fund was established to provide assistance for short-term improvements at RBMK reactors, considered one of the higher risk designs, and at older light-water reactors, such as the VVER-440/230s. He noted that, to date, Nuclear Safety Account grants to Bulgaria and Lithuania have been conditional—that is, the recipient countries have agreed to an early shutdown of a higher risk plant in exchange for financial assistance to upgrade the safety systems of a more modern plant.

Operational safety assistance has been provided to 13 plants through operator and fire safety training and equipment and through symptom-based operating procedures. These plants serve as models that the host countries can use for improving safety at other plants. In addition, improved equipment and training have been provided to their regulatory authorities. These efforts have been coordinated by the Department of State and have been conducted primarily by the Department of Energy and the Nuclear Regulatory Commission with funds and support provided through the U.S. Agency for International Development.

Dr. Lash stressed that the United States, together with its G-7 and G-24 partners, has encouraged the shutdown of higher risk plants in Central and Eastern Europe and the New Independent States while also providing safety improvements at operating nuclear power plants.

There are lessons to be learned from our recent experience in addressing the safety problems of Soviet-designed nuclear plants. Among these lessons is a better understanding of the complex problems faced by the nations we are attempting to help—problems that are an innate part of the social, political, economic, and cultural realities left by the Soviet-style regimes that created them. The countries of Central and Eastern Europe and the New Independent States are faced with difficult choices in the efforts to improve the safety of their nuclear power plants.

The fundamental challenge for the United States is to work cooperatively to minimize the risks of all nuclear power plants. As we address the most critical elements of the nuclear safety problem, including early closure of the riskiest plants in Central and Eastern Europe and the New Independent States, we should also seek to use our resources in ways that build up the indigenous capabilities in each country for improving the safety of its own nuclear power plants. As we work together to strengthen the nuclear safety infrastructure, we can help instill in designers, operators, management, and regulators the safety culture that is prevalent in the older democratic countries. In the longer term, nuclear power plant safety will be dependent on the safety infrastructures and cultures in the host countries.

Dr. Lash also said that one of the complications that has arisen in the United States' efforts to address safety problems is the issue of liability. For a period of time this past year, the liability issue delayed the provision of some hardware for safety system upgrades in Russia and Ukraine. Dr. Lash reported that the bilateral agreements concluded with Russia and Ukraine provide appropriate protection from nuclear liability for some qualified companies under contract to the U.S. Government. This provision has enabled U.S. Government-funded work to move forward at Soviet-designed plants in these two countries. He pointed out that we still urgently need to address long-term solutions to the liability issue. He asserted that countries with operating Soviet-designed nuclear power plants must be a part of strong national and international nuclear liability regimes.

Further, Dr. Lash noted that adopting domestic liability legislation and joining an international liability convention would provide to the host countries much-needed benefits, such as a free flow of nuclear safety equipment and services under bilateral programs. Most importantly, a satisfactory liability regime would allow firms within the United States and other countries to undertake work on a commercial basis that would improve the level of nuclear safety in Ukraine and Russia.

He argued that adequate funding for materials and equipment and for highly qualified staff is essential for the development of a strong safety infrastructure and culture. Western nuclear safety infrastructures are supported by revenues from the sale of electricity. Market reform in host countries, therefore, will be necessary to ensure that nuclear power plant operators are paid for the electricity they produce so that they have the funds needed to maintain and operate their plants safely.

Dr. Lash concluded by saying that the United States intends to continue our close cooperation with countries that operate Soviet-designed reactors and, where possible, to fund programs that substantially contribute to the host countries' efforts to reduce the risks of their nuclear power plants.

He commended those U.S. utilities and the World Association of Nuclear Operators (WANO) which have made significant contributions to our international safety goals by hosting visiting operators under various "twinning" programs.

PLENARY SESSIONS

Dr. A. David Rossin was the rapporteur for the Plenary Sessions. He observed that throughout these sessions the matter of liability kept coming up. He remarked that the issue is one of prudence. The accidents of concern are low-probability events; the business risk comes from the fact that, if a lawsuit is filed, the entire assets of a contracting company could be in jeopardy. Simple business logic says that it is not prudent to risk the whole entity for one project, perhaps of marginal profit potential, even though its purpose is noble, and there might be some real business somewhere out in the future.

Several speakers said that if Russia and Ukraine would join the Vienna or Paris Conventions the liability issue would be resolved. The problem is that this step has not happened, time is passing, and the world is concerned about safety. Dr. Rossin argued that where safety is at stake action should be taken without undue delay.

He also noted that contractors from other nations are constrained by the liability issue as well, but the issue is perhaps more sensitive in the United States because of our history of lawsuits where liability is not strictly limited and the issue is emotional, such as with radiation, tobacco, or any potential cancer-causing agent.

Dr. Rossin further observed that Eastern European nations have made their own determination that the downside risks of closing plants are greater than the risks of continuing to operate. These nations have made a "How safe is safe enough?" determination driven by a cost–risk analysis, which includes the stark realization that they do not now have the economic resources to pay the costs of eliminating all the risks they might wish to deal with.

Money to pay operators and regulators is not in hand. Several speakers noted that raising electric rates is not the only matter; rather, electric bills from factories and other state-owned enterprises as well as from individuals remain unpaid. Also, even if enough money is collected from rates, the government only passes on a portion of the collected money to the electric company to meet its costs. No money is available to set aside for plant improvements or investment in alternative capacity or even to fund the regulatory body.

Dr. Rossin asserted that the most challenging concept advanced during the Plenary Sessions was the desperate need for developing a safety culture at the highest level. This challenge is different from what was discussed in terms of operators and managers. This challenge involves building an understanding of what safety means, learning that safety in the future is never absolutely sure, understanding that things do go wrong and accidents can happen, and finally realizing that dedicated people can and will work hard to reduce risks to a practical minimum.

He also remarked that this challenge means understanding that independent safety regulators are essential and that they must have the authority to act as well as the experience and courage to act responsibly. Strong regulation does not mean only having the authority to stop operations to penalize operators or to look powerful. This authority also means the strength to make technically sound and responsible decisions in the face of pressures from all corners, which is the way to build mutual respect.

WORKING SESSION I: RBMKs

The session began with introductions from E. Ivanov, Rosenergoatom, and Professor J. Vilemas, Director of the Lithuanian Energy Institute. The rapporteur for this session was M. Hayns of the United Kingdom. Although both Ivanov and Vilemas described the difficult position currently being faced by the operators of these plants, Ivanov—in particular—was very positive concerning the plants' future operation. Vilemas emphasized the importance of supporting not only the plant and its operational needs but also the regulators and the other infrastructure that goes to underpin the safe operation of the plants.

The discussion focused initially on the criteria that would be applied in considering whether to shut down any particular plant. Mr. Hayns observed a strong feeling on the Eastern side that strong statements were still being made about the safety of the RBMK designs, even though the West should now clearly understand that there was a range of plants of different stages of development. Perhaps, unlike the West, a number of very complex economic, sociopolitical, and technical factors had to be balanced before any such discussions could be held. Furthermore, considerable regional differences also existed. Some persons felt that the G-7 calls for closure were based more on political than technical issues. In this context, Mr. Hayns noted that the speech given by Dr. Kopchinsky (although not in this session) concerning the status of the Chornobyl reactors really emphasized the extreme difficulties facing discussions over the future of the plants. Nevertheless, he was very clear and explicit in his criticism of previous governments in the Ukraine for their lack of attention to nuclear safety and for their vacillation over the future of these plants.

In response to a question, Dr. Ivanov wished to make clear that he foresaw no requirement for long-term assistance from the West but rather would wish to see the current activity as temporary and as leading to a long-term relationship of collaboration and cooperation. This statement also reflects the views of Dr. Ponamarev-Stepnoi in having a very positive view of the future of the nuclear power industry in the Russian Federation. He was confidently predicting a thriving export market for their newer designs.

Mr. Hayns remarked in response to a question that there was a lively discussion on the *positive* aspects of the RBMK design. This positive discussion was considered to be a useful counterbalance to the generally negative statements about the "Chornobyl-type" reactors. Eric Sodermann, leader of the probabilistic safety analyses (PSA) performed under the Barselina project, supported the arguments. The particular aspects seen as beneficial were the very high thermal capacity of the core, the high coolant inventory, and the difficulty in generating "whole core" accidents. In addition, the operational advantages of on-line refueling and, in principle, low radiation doses were seen as beneficial. Although not discussed in detail, the question of multitube ruptures also arose. Clearly some difficulty still exists in establishing a credible model that can be used to underpin the safety case for such an event.

One of the underlying themes of the whole seminar had been the question of improving the "Safety Culture" of nuclear power plant operations. In his talk on the situation in the Ukraine, Nikolai Steinberg had highlighted this question as probably the highest priority item. This discussion was thought to have special significance for the Chornobyl plant, where the uncertainty surrounding future operations and the poor working conditions did not lead to an environment conducive to safe operation.

Finally, as Mr. Hayns reported, the discussion period ended with a more technical debate focused on codes and data for the calculation of neutronic and thermal–hydraulic conditions in the core. Alan Brown [Atomic Energy of Canada Ltd. (AECL)] reported that calculations using Western codes had given a reassuring verification of the Eastern calculations. Further work was under way, and not until early 1996 will a full verification of the calculations for the Ignalina safety analysis report be available. The question of transient calculations was still open, however, and further work was needed to establish the credibility of the available codes. A problem was raised concerning the availability of consistent data. This problem was causing real difficulties for Western analysts trying to make calculations on these plants. Unfortunately, no clear solution emerged, and this area requires continuing efforts to resolve.

WORKING SESSION II: VVER-230

The rapporteur for this session was Dr. A. Birkhofer of Germany. Dr. Birkhofer noted that the discussions indicated the operational safety of VVER-230 reactors has improved over the last years. In this respect, the measures for Kozloduy within the PHARE program of the European Union, the assistance program through WANO, and the ongoing improvement programs for Bohunice as well as for Kola and Novovoronesh were mentioned. The improvements covered updating and completing operating procedures (for example, fire protection and fire fighting as well as in-service inspection).

Dr. Birkhofer observed that a major problem still under consideration for all VVER reactors and particularly for the older type-230 reactor is embrittlement of vessel material and, especially, the measures to reduce the possibility of thermal shocks. The integrity of the pressure vessel is of prime importance for the safety of all nuclear power plants. Therefore annealing has been performed for all vessels reaching excessively high ductile-to-brittle transition temperatures. He asserted that uncertainties still existing should be removed by further investigations and analyses, taking into account sufficient experimental evidence regarding fracture resistance of the material. He suggested that verifying the completeness of the relevant low spectrum used in the analyses would be desirable and also checking whether the most critical crack-initiating phenomena have been considered. Cold overpressurization of the whole system and transients caused by a loss-of-coolant accident (LOCA) or steam-line break with cold-leg injection are of particular concern in this regard. Problems to be solved include the following:

- The validity of large variations from plant to plant in the initial transition temperature and the establishment of a conservative upper bound for this value.
- The verification of homogeneity of the weld material through the vessel wall.
- The accuracy of the empirical model of transition temperature drift as a function of neutron flux.
- The reembrittlement and the effectiveness of multiple annealings.

The first part of the discussion dealt with the reduction of vulnerability from accidents and with improvements in defense in depth, whereas the second part dealt with accident mitigation systems and measures. Because of the large conservatism in former loss-of-coolant analyses, best-estimate calculations indicate that a LOCA up to a 200-mm-diameter break (surge line) can be covered by existing emergency cooling systems. In discussions on limiting LOCA, a suggestion was made to reevaluate whether filtered containment venting should deserve higher priority for backfitting. Furthermore, one person mentioned that accident-management procedures should be more thoroughly investigated, taking into account the large thermal inertia of VVER-230 plants.

Dr. Birkhofer pointed out that PSA has been considered an appropriate tool to indicate the safety level of the plants; however, there was agreement also among the present representatives of licensing authorities that those values should not be directly used for licensing decisions. They could, however, be used, for example, to decide on alternative-upgrading measures.

Comments were also made on the G-7 conclusions, especially in view of VVER-230 reactors. The discussions since the Munich summit did not reflect the progress that has been made or that is planned at various plants. In this respect, all representatives from licensing authorities or nuclear operators of the three countries operating VVER-230 reactors indicated that no long-term licenses exist for those plants. Rather, for each unit, the operation is based on an annual permit that reflects the operating history and the improvements that have been made in the preceding years. The permit also lays down refurbishment measures for the forthcoming year.

In this regard, various representatives from Eastern European organizations suggested a common evaluation of how and why older western plants have been refurbished in the past. Dr. Birkhofer noted that such an exercise would help to understand why and to what extent safety improvements were necessary and how those improvements have been performed.

WORKING SESSION III: VVER-440/213

The rapporteur for this session was Dr. M. H. Fontana of the United States. He reported that the VVER-440/213 plants have been extensively reviewed by internal groups and have been reviewed externally by the German GRS, International Atomic Energy

Agency (IAEA) OSART, IAEA ASSET, and WANO. Significant experience with these plants exists, as evidenced by the number of units in operation: Czech Republic has 4; Hungary has 4; Slovak Republic has 2 plus 4 under construction; Russia has 2; and Ukraine has 2. Information sharing occurs through user's groups and by informal information transfer.

The Hungarian and Czech participants indicated that validating safety as being equivalent or better than western plants is very important before joining the European Union by the year 2000.

The VVER-440/213 designs are significantly different from Western designs, particularly with respect to the horizontal steam generators and the multiple-tray pressure-suppression system instead of a pressure-tight containment system. The VVER-440/213 designs have positive safety features.

- They have a significant level of inherent safety because of the following:
  - A large coolant inventory/power ratio. This large ratio results in slow transients and more time available for corrective action if appropriate. (A fire in Armenia showed that the plant withstood over 6 h without cooling with no fuel failures.)
  - Conservative fuel design.
  - Conservative pressures and temperatures.
  - A small core; therefore Xenon oscillations are not a problem.
  - Multiple loops.
- The design-basis accident is the large-break LOCA. In this respect, the plant is equivalent to western plants; however, questions would exist regarding the capability of the plant to withstand beyond-design-basis accidents.
- The confinement/pressure-suppression system is designed to withstand the design-basis accident but needs to be further proved.

Dr. Fontana observed that the steam-generator performance of the Hungarian plants has been excellent. In 40 reactor-years of experience with four units, each having six steam generators with 5536 tubes each, only 20 tubes have needed to be plugged.

Extensive safety analyses have been performed for these plants. At Dukovany, all the IAEA safety issues have been reviewed and judgments made as to relevance to Dukovany. Those issues which are relevant have been placed in priority order for further assessment or development. A Level I PSA has been performed, which gives a core damage frequency of 3 to 9×10^{-5}. (PSA issues were discussed in another breakout session.)

Dukovany's modernization goal is to be safe to its end of life if plant life extension will allow operation for 40 years. The target is to show that the plants are as safe as Western plants of the same age after the year 2000.

Important safety issues, for example, for the Dukovany plant include the following:

- Instrumentation and control replacement (approved for implementation).
- Steam and feedwater line integrity.
- Equipment qualification.
- Extension of the leak-before-break analytical model to provide support for bubble condenser behavior.
- Prevention of pressurized cold thermal shock to the reactor pressure vessel.
- Primary cooling circuit cold overpressure protection.
- Reliability of the diesel generators.
- Bubble condenser behavior.
- Internal hazards caused by high-energy pipe breaks.
- Training with a full-scale simulator.

Highest priority safety upgrading for the Paks plant includes the following:

- Implementing complete separation of the emergency feedwater system from the normal and auxiliary feedwater.
- Reducing human errors during operation and maintenance.
- Decreasing the consequences of leakage from the primary to the secondary system through steam-generator tube ruptures.
- Installing a hydrogen removal system in the containment for design-basis accidents.
- Preventing sump clogging.
- Reducing the probability of failure in the emergency core cooling systems in switchover to the recirculating mode of operation.

Other important tasks for the PAKS plant include the following:

- Developing primary and secondary feed-and-bleed procedures.
- Protecting the reactor pressure vessel against overpressurization in cooled-down situations.
- Avoiding (boron) diluted water in the core during shutdown.
- Avoiding dropping heavy loads from cranes.
- Increasing the reliability of the high-pressure injection system.

An investment banker in the workshop made the comment that, for significant Western investment to occur, the safety issues must be made understandable to the average investor.

In conclusion, Dr. Fontana said that much thought and effort obviously have gone into the design, operation, and upgrading of these plants. The plants have certain inherent advantages because of their high water-to-power ratios, but the effectiveness of their bubble confinement system needs demonstration, research, and development. PSAs have been used to provide guidance on operations and upgrades. Extensive use of outside reviews and rigorous assessments of the IAEA list of safety issues as they apply to these plants have been made. On the basis of the workshop discussions, the staffs of these plants appear to be well on their way to their goal of becoming equivalent to or better than Western plants by the year 2000.

WORKING SESSION IV: VVER-1000 REACTORS

The rapporteur for this session was Dr. W. Horak of the United States. He remarked that VVER-1000 reactors, originally designed in the 1970s, were designed in accordance with the existing Soviet standards, OPB-73 and RPB-74. The later serial production model, V320, was designed in accordance with OPB-82. All VVER-1000 models are designed for double-ended pipe breaks, single failures, and defense in depth. Dr. Horak observed that, in general, the level of safety has increased with each new generation of VVER-1000. One of the newest designs, the VVER-91, is designed for a 4 × 100% systems capability.

The VVER-1000 design has many strengths:

- 3 × 100% capability for safety systems.
- Physical separation of safety systems.
- Prestressed-concrete containment buildings with steel liners.
- Containment leak tightness that has been measured to be within Western standards.
- Systems to cope with internal and external hazards.

According to Dr. Horak, Rosenergoatom has begun a two-phase upgrade for all operating VVER-1000 reactors. The upgrade program is making use of a Level 1 PSA done for the Balakova-I plant. The PSA has identified station blackout transients as a major contributor to the core melt frequency. To improve the station blackout performance, the upgrade program is increasing the emergency feedwater supply and replacing the steam-generator relief valves. These changes will allow for 8 h of cooling until steam-generator dryout. Other upgrades are being done to address beyond-design-basis accidents and anticipated transients without scram. Quality assurance programs have been developed and are being applied throughout the upgrade program.

In Ukraine, a similar upgrade program is being followed. The Zaparozhye V and VI Units, however, because of their recent construction, contain many of the planned improvements. Of equal importance to hardware improvements in Ukraine is the improvement of training and certification programs. The old training methods no longer apply. Recently, a full-scope simulator, built by S-3 and VNIIAES, was installed. Certified maintenance training programs have also been initiated.

An important part of the upgrade program has been the improvement of diesel-generator performance. The program, intended to reduce the start-up time of the diesel generators, has been completed at the Kalinin plant. Problems have also been experienced with the diesel generators at the Kozloduy plant.

Finally, Dr. Horak noted that a need exists for coupled three-dimensional neutronics–thermal–hydraulic computer codes. Such codes have been used extensively in Finland, not only in support of Loviisa but also for VVER-1000 analysis.

WORKING SESSION V: REGULATION, STANDARDS, CRITERIA

The rapporteur for this session was A. Carnino of the IAEA. She reported that the session was devoted mainly to the presentation of the regulations used in the former Union of Soviet Socialist Republics (USSR) and subsequently just in Russia for nuclear power plants.

Mr. Bukrinsky of the Russian regulatory organization GAN discussed the development of regulations; the first period was based on industrial standards supplemented by radiation protection, core physics, and metallurgy norms (first generators of VVER-440 and RBMK-1000). In 1973, the approval of OPB-73 started with the creation of safety emergency cooling and localization of safety systems, designed for the credible design-basis accident (VVER-440, RBMK-1000 of the second generation, and VVER-1000). In 1982, OPB-73 was revised and led to OPB-82. After the Chornobyl accident, the document was further revised, and in 1988 this revision became OPB-88, which includes the review of beyond-design-basis accidents with possible severe core damage. This document has been effective since July 1, 1990. Following this development was the creation of an independent state body for the regulation and supervision of nuclear safety.

Mr. Stuller of the Czech Republic indicated that a new nuclear act was developed and has been submitted for parliamentary approval. The Czech plants are based on the former USSR OPBs. Ms. Carnino noted that one major problem encountered today is the development of industrial standards as used in Western countries but still reflecting what was used on existing plant structures, materials, and components. Such an example is nondestructive testing.

The IAEA worked together with Gosatomnadzor. [Mr. Bukrinsky has developed a comparison between OPB-88 and the NUSS IAEA Safety Series documents (codes and guides).] The major findings are given in the following discussion.

The Russian safety concept, as reflected in OPB-88 and PBJa89, is comparable to NUSS; however, differences occur both in the approach and in details, especially with respect to the following points:

- Classification of OPB-88
  - Class 1 only includes the pressure vessel and the fuel elements. The reactor coolant pressure boundary is Class 2 and Class 3 and includes all elements of redundant safety systems.
  - The classification is for the elements. Consequently safety systems can be composed of elements belonging to different safety classes.
  - A wider range of normal operating systems is considered to be important to safety.
- Reactivity coefficient
  - PBJa89 requires that all the individual reactivity coefficients be negative in the entire range of reactor parameter variation.
- Containment venting
  - According to OPB-88, during severe accidents, venting is permitted through a filter without isolation devices.
- Criterion for excluding evacuation
  - According to OPB-88, to exclude evacuation, the probability of an unacceptable event (evacuation of population centers with more than 100 000 persons) for future plants is required to be less than 10^{-7} per reactor year.
- Anticipated transients without scram (ATWS)
  - ATWS are not considered in the Russian documents for existing plants. In the future, ATWS will be considered as beyond-design-basis accidents.
- Coverings and coatings
  - Requirements do not exist for coverings and coatings for components and structures within the containment. This list of requirements could lead to a degradation of other safety functions (for example, cooling capability) or to corrosion of components of the safety systems.
- Analysis of severe accidents
  - In considering beyond-design-basis accidents, requirements do not exist with respect to realistic analyses (best estimate) and representative dominant severe accidents.
- Design requirements for the containment
  - Requirements for the containment are only contained in Report No. PNAEG-10-21-90, which came into effect in 1991. Consequently these requirements were not fully applied to existing plants.
- Responsibility for safety
  - OPB-88 does not clearly define the responsibility of the operating organization for nuclear safety and the delegation of authority to the nuclear power plant management. Two operating organizations—Rosenergoatom and Leningrad nuclear power plant—have been nominated, but this nomination is still subject to approval by Gosatomnadzor of Russia.

This report addresses the first objective of the comparison (that is, to find possible differences between the safety concept of the Russian documents OPN-88, PBJa-89, and the NUSS documents 50-C-D, 50-C-O, 50-SG-D11, and 50-SG-D14); however, the existing nuclear power plants with VVER and RBMK reactors have been designed in accordance with OPB-73 and OPB-82. OPB-82 was the result of a revision of OPB-73 on the basis of the experience gained, but the underlying safety concept has remained practically unchanged. Ms. Carnino concluded that, for a better understanding of the safety of existing nuclear VVER and RBMK plants, especially with respect to their design, a comparison of the safety concept of OPB-82 with the NUSS requirements is essential; this comparison work is planned as the next step.

WORKING SESSION VI: EXPERIENCE OF WESTERN CONTRACTORS

The following discussion is based on the principal points that were summarized and presented by rapporteur D. Squarer.

The session was attended by various organizations, including large vendors and contractors (e.g., EdF, Westinghouse, and General Atomics); architect and engineering firms (e.g., Bechtel, Gilbert, and Raytheon); national laboratories (e.g., Brookhaven National Laboratory, Argonne National Laboratory, Idaho National Engineering Laboratory, and Pacific Northwest Laboratories); U.S. Department of Energy; U.S. Department of State; and small- and medium-size contractors (e.g., EQE, Science Applications International Corporation, Viking, MPR, and Sorrento Electronics).

The session co-chairs were Mr. J. Baret of France and Mr. P. Yanev of the United States. Views and opinions from Eastern Europe were expressed by Dr. J. Gado of the Hungarian Academy of Science, KEIKI Atomic Energy Research Institute, and by Dr. Y. Yanev, Chairman of the Committee on the Use of Atomic Energy of Sofia, Bulgaria, both of whom actively participated in the session and injected realistic perspectives from Eastern Europe.

Dr. Squarer remarked that Mr. P. Yanev has been operating in Bulgaria for several years and has established an office in Bulgaria that is staffed with local personnel. His experience and the experience of others is that, although we are witnessing a transition to capitalism in Eastern Europe, making a profit is still difficult. This issue is significant because the Eastern Europeans perceive that the first priority of the Western contractors is to make a profit, whereas the reality is such that, in spite of the relatively inexpensive labor, operating profitably in Eastern Europe is difficult.

Dr. Squarer also observed that EdF has operated in Eastern Europe for some time. The organization functions more as a consultant to nuclear operators (that is, the utilities), and it seeks to promote collaboration (twinning) with these utilities and has worked closely with WANO. Its technical know-how areas of expertise include architect–engineer, vendor, and operator of nuclear power plants.

Dr. Squarer noted that both a perception and ample evidence exist that all the contractors are overburdened with substantial bureaucracy on both sides: the Western governments, as well as the Eastern European governments and institutions. As an example, the U.S. bureaucracy involves the White House (G-7 and the Lisbon Initiative), U.S. Department of Energy, U.S. Nuclear Regulatory Commission, U.S. Department of State, IMF, World Bank, several U.S. national laboratories, and possibly others. He remarked that this situation must be improved and simplified for Western contractors to be able to operate in the spirit of a free market economy.

Other remarks included the following:

- The Eastern Europeans have a tendency to dwell on technical details, even if the schedule cannot be met.
- Meeting the schedule of the safety upgrades of Soviet-designed reactors is an important aspect that cannot be overlooked.
- Western contractors who wish to operate in Eastern Europe should be ready to invest their own funding initially.

Dr. Squarer reported that foreign aid often distorts the marketplace by raising the local wages and the expectations disproportionately to the overall change in the standard of living. This distortion, in turn, may further erode the profit margin of Western contractors and may instigate a disincentive to operate in Eastern Europe. He suggested that an alternative approach may be to pay directly to the end users (that is, to the owner–utility of a particular plant).

The following question was raised: How long should the assistance program last? In terms of the safety of Soviet-designed reactors, the answer may be until the safety of the reactors is at par with the West and a permanent regulatory and industrial infrastructure is in place that will ensure the continued safe operation of the reactors according to international standards and the safe design and construction of newly designed and constructed reactors.

Dr. Squarer reported an obvious lack of coordination between different Western contractors who work at a single Soviet-designed reactor site. The site assumes and looks for the Western contractors to coordinate their work; however, no such coordination exists at present. This situation exists because of complicated government-to-government funding arrangements, and the problem may have a direct bearing on the outcome of the safety upgrades. He noted that it could be resolved on a site-by-site basis if the hosting site will assume the responsibility of coordination.

Dr. Squarer reported that the issue of liability of Western contractors performing work in Eastern Europe is still an important consideration; however, liability does not appear to be the major impediment to perform work and to deliver equipment to Soviet-designed reactor sites because some liability insurance has been assumed by Western governments and some liability insurance is carried by the Western contractors themselves. If liability was a formidable issue, then no work would have been performed on Soviet-designed reactors, which is not the case.

Dr. Squarer observed that the perspective of the Eastern European is that the first priority of the Western contractors is to make a profit, and upgrading the safety of the reactors is of lower priority. Because perception can often be changed by better communication, improving the communication between the Western contractors and the Eastern Europeans is desirable. This concern will likely be diminished as the Eastern European society continues in its conversion to a capitalistic society.

Getting paid for work done on upgrading the safety of Soviet-designed reactors in Eastern Europe is a major issue. The Eastern European sites do not have funds of their own and can barely survive, and very few Western contractors are willing to receive payment in local currencies. Government funding is typically not allocated to the site or to the utility where the actual work is to be done. Government funding thus far has been allocated mostly to (paper) studies rather than to the supply of actual hardware equipment or to fund directly a specific site for the purpose of purchasing equipment; however, the trend appears to be changing from "studies" to "equipment" now.

According to Dr. Squarer, the private sector needs to be more involved in the safety upgrade programs. Government agencies have had an important role in laying the foundations for these programs, arranging for funding, minimizing the liability to the contractors, and establishing credible regulatory organizations. Now the government agencies should help in a smooth transition to allow the private sector to operate because private companies must execute the actual work on site. This transition can be accomplished by cooperation between government, plant operators, regulators, and private industry. Also, an infrastructure must be developed in Eastern Europe that will serve the need of Western contractors. This need is currently being fulfilled by the individual contractors.

Dr. Squarer finally remarked that we could consider the concept of "conservation of risk" in an analogous manner to the concept of conservation of energy, momentum, and mass. Risk cannot always be eliminated; rather, it is often *shifted* to another sector of the population or to another site. When radioactive waste is transported from a plant and buried at a different site, the risk is shifted to the burial site. When a nuclear power plant is shut down, the reduced risk at the plant is shifted to the population at large, which is put at risk if no replacement power is provided.

WORKING SESSION VII: TRAINING, PROCEDURES, OPERATIONAL SAFETY

The rapporteur for this session was Professor C. Heising of the United States. She reported that three presentations introduced the topic and described its status. Dr. Sonja Haber (Brookhaven National Laboratory) discussed training accomplishments of the U.S.–Lisbon Initiative contributions, which included the following:

1. Application of the Systematic Approach to Training (SAT) to develop pilot programs in 12 areas analogous to Institute for Nuclear Power Operations (INPO) programs.
2. Delivery of two such training courses and special courses including safety culture.
3. Delivery of infrastructure items such as computers and copiers for training centers.
4. A soon-to-be announced award for supply of a full-scope simulator.

A. Kroshilin spoke of Russian training programs and specifically identified the following:

1. The education level of Russian operators is high, and, in fact, their academic requirements exceed those in the United States.
2. A symptom-oriented procedure development approach is in use.
3. All stations now have training centers of varying levels of complexity, and two stations (Balakova, Zaparozhe) have full-scope simulators.

Sandy Hastie of WANO (INPO) provided an overview of WANO and its program, including:

1. Operations experience exchange.
2. Operator-to-operator exchange.
3. Good practice sharing.
4. Performance indicator definition and reporting.
5. A new voluntary peer review program.

Professor Heising noted that the points and issues in the subsequent discussion included the following:

1. Agreement on the value of external peer review, today exemplified by the WANO and IAEA–OSART programs.
2. Emergency planning and response capacity and the role of training (here it was noted that negotiations are under way to upgrade a Russian crisis center).
3. The effectiveness of training both in terms of its quantification and acceptance (here it was noted that operators will follow procedures that they appreciate and the basis of which is understood).
4. The institutional framework and the relationship between training and regulation; operators prefer independence to facilitate free information exchange and performance improvement, whereas regulators would like access to performance indicators and documentation that plant event experience is disseminated and utilized.

WORKING SESSION VIII: PRA RESULTS AND APPLICATIONS TO SAFETY IMPROVEMENT

The rapporteur for this session was J. Bickel of the United States. Mr. Bickel noted that, in the time period since the First ANS Special Workshop on the Safety of Soviet-Designed Reactors, significant development of level one Probabilistic Risk Assessments (PRAs) has occurred. Level one PRAs are well under way, or now exist, for the following plants:

Ignalina Unit 1 (RBMK)	Lithuania
Bohunice Unit V1 (VVER-440 Model 230)	Slovakia
Dukovany Unit 1 (VVER-440 Model 213)	Czech Republic
Paks Unit 1 (VVER-440 Model 213)	Hungary
Kozludoy Unit 5 (VVER-1000)	Bulgaria
Temelin Unit 1 (VVER-1000)	Czech Republic
Rovno Unit 1 (VVER-440)	Ukraine

This list represents a significant accomplishment by all the parties involved.

The PRAs were undertaken for a variety of reasons in each of the countries in question; for example, the PRA for Bohunice was undertaken to characterize the existing levels of risk and to quantify the magnitude of improvements to be obtained by carrying out individual projects that are a part of the current major reconstruction effort. The PRA for the Paks plant in Hungary was part of an overall periodic major safety assessment of the units. The PRA being performed for Temelin in the Czech Republic is being used to provide additional information in the ongoing licensing effort. The Ignalina PRA, performed under the Barselina Project as a bilateral effort between Sweden and Lithuania, was performed to train plant personnel and regulators on the major sources of risk at the plant.

Mr. Bickel remarked that the efforts of the members of IAEA's VVER-PRA Working Group to deal with common issues, such as initiating events data and component reliability, was noteworthy. IAEA obtained participation from each of the countries operating VVERs as well as the design institutes involved in the original development of the plants.

A comment was made by Dr. Robert Budnitz that the level one internal events PRAs are useful but they will likely capture only half the risk of these plants. The United States and West European experience indicates that more than 50% of the risk of nuclear power plants comes from external events. As one future effort, evaluating the risks posed by seismic events and fires would be desirable. Carrying out such external events analysis, however, will be hampered by lack of detailed records of (1) how electrical cable was actually routed in the plants and (2) whether all structural elements called for in design specifications were ultimately installed. This recognition may necessitate some type of simplified analysis approach. Seismic walkdowns to identify possible seismic outliners might help reduce or eliminate certain problems, but these walkdowns make assumptions on the integrity of buried or encased elements that cannot be seen.

The level of detail and sophistication among the various PRAs was also discussed. The Paks PRA was unique in its use of simulator experiments to obtain information on operator performance during severe transients and accidents.

Mr. Bickel concluded that, in the future, a number of the PRAs will probably be expanded to level two and higher to better understand the differences in containment/confinement performance.

WORKING SESSION IX: CONTAINMENT/ CONFINEMENT ENHANCEMENTS

The rapporteur for this session was Dr. W. Deitrich of the United States. Mr. Misak of Slovakia opened the session with a discussion of progress on confinement improvements on the VVER-440/230 and 440/213 nuclear power plants in Slovakia. Slovakia operates both plant types on the Bohunice site, which hosts two VVER-440/230s and two VVER-440/213s with bubbler tower confinement pressure suppression.

The Bohunice VVER-440/230 reactors have a confinement volume of 10 000 to 12 000 m^3 with a design pressure of 0.1 MPa. Blowoff flaps protect against confinement overpressure for breaks up to 100-mm equivalent diameter by venting to the atmosphere. A spray system is provided to prevent flap opening for breaks up to 40 mm in diameter. The confinement design provides poor control of releases with some release even for small accidents (>40-mm breaks) through the blowoff flaps. Structural integrity would be preserved for larger breaks. The estimated confinement leak rate was 5000% per day prior to upgrades.

Confinement improvements have reduced the confinement leak rate to about 300% per day and have improved the reliability of the spray system. The pressure capability of the confinement structure has been examined with the conclusion that integrity can be maintained up to 0.13 MPa. The ultimate overpressure capability is estimated to be 0.15 MPa.

Regulatory requirements for confinements have been established, including the following:

1. Demonstrate the limiting confinement pressure capability.
2. Show the capability to cope with a 200-mm break without exceeding the limited consequences.
3. Show reliable opening and closing of confinement vent flaps.

Best-estimate methods are to be used in making confinement performance estimates.

Mr. Misak continued to discuss the VVER-440/213 confinement system. Its volume is larger than that of the VVER-440/230s, at 15 000 m^3, with a pressure rating of 0.15 MPa. The VVER-440/213 reactors have bubbler condenser towers to reduce releases. Issues to be addressed with the VVER-440/213 confinement include the following:

1. Transient performance of the bubbler condenser water trays and steel supporting structures.
2. Vibration resources between steam flow and structures for small-break LOCAs.

Mr. Misak suggested that full-scale tests of the bubbler condenser tower performance are needed.

Mr. Koshmanov reported on the ongoing confinement improvements in the two VVER-440/230 units at the Kola Nuclear Power Plant in Russia. Like the Slovakian VVER-440/230s, the confinement structures in the Kola units do not meet modern leak standards. Improvements now under way should reduce the leak rate to about 100% per day. The pressure capability is estimated to be 0.15 MPa, but more ventilation and recirculation flaps are needed to guard against overpressure for large breaks.

The Kola confinement leak-rate improvement work is a success for the Lisbon Nuclear Safety Initiative (LNSI) program. Sealing technology from the United States has been applied to sealing of cable penetrations, valve stem penetrations, and weld seams. Also, the United States provided sealing material and application technology for one unit with training of Russian personnel and application equipment to facilitate use of other units. In addition, rubber seals on confinement doors and vent flaps have been repaired to reduce leakage.

Mr. Koshmanov further commented on the VVER-1000 containment capabilities. The Balakova containments are rated for 4 kg/cm^2 pressure with a leak rate of 0.3% per day. This plant is pursuing reduction in testing time and is installing state-of-the-art leak measurement and detection equipment.

Mr. Gennady Zeltobriuck of Lithuania reported on confinement improvements at Ignalina. These units are designed for 3 kg/cm^2 overpressure with a 56 000-m^3 volume. They can withstand a 900-mm break while maintaining integrity. After the Chornobyl accident, a decision was made to reduce the leak rate on the basis of a criterion related to the thyroid dose to a baby in the "sanitary protection zone" surrounding the plant. This criterion can be met if the plant output is limited to 1250 MW. The annual leak tests were described as a "challenge."

Someone also noted that badly needed ultrasonic test equipment for Ignalina is still stranded in Sweden because of liability concerns.

Dr. Deitrich remarked that, in general discussion, the subject of improved reactor cavity venting capability in RBMK reactors was addressed. The objective is to increase the number of simultaneous pressure tube failures that can be sustained without lifting the upper shield plate. Space exists to install venting equipment, including a 600-mm outlet pipe, to vent flow from up to nine simultaneous failures from the reactor cavity to the confinement volume. Projects to install such pipes are under way at Leningrad (Sosnovy Bor), Smolensk, Kursk, and Ignalina. Concern was expressed over dynamic stability of various structures under flow conditions associated with such venting. Experiments and some structural strengthening were stated to be required. Dynamic behavior of bubbler condensers is in question in this case, as it is in the VVER-440/213s.

Dr. Deitrich noted that filtered vented containment was discussed. Filtered ventilation for VVER-440, 230s and -440/213s has been proposed (as part of the LNSI program plan) but is not currently being pursued. The concept is to provide forced postaccident confinement ventilation to maintain negative confinement pressure and reduce release caused by leakage. Consideration is being given to filtered venting of VVER-1000 containments in the future to accommodate severe accident loads for which the structures are not designed at present.

Finally, the need to use physical assumptions in design of containment/confinement modifications was discussed. Best-estimate analysis, rather than bounding assumptions, is needed.

WORKING SESSION X: FIRE PROTECTION/FIRE FIGHTING

The rapporteur for this session was Dr. M. Levenson of the United States. The session was co-chaired by E. G. Diatian of the Ukraine and E. S. Ivanov from Russia. The co-chairmen set the tone of the session by saying that, although the current fire protection standards set by the Soviet Union in 1988 were quite acceptable and similar to current international standards, most of the Soviet-designed plants were built prior to 1988 and have not yet been backfit or upgraded. As a result of this opening, the session was not handicapped by discussions as to whether or not improvements were needed. Dr. Levenson noted that the workshop was free to focus on issues of what to do, how to do it, and how to select priorities. The priority issue occupied a major fraction of the time and included some rather heated discussions among the participants, including the co-chairmen. Dr. Levenson remarked that the issue in question was how to ensure that the objective would be achieved.

The starting framework for the workshop was that fire protection consisted of three different issues or areas of concern: (1) fire prevention, (2) detection and localization of fires not prevented, and (3) fire fighting. All three areas need improvement and should be addressed.

According to Dr. Levenson, the active discussion started over a dissenting view that safe shutdown and post-fire control were not included in the basic format. By the end of the session, agreement was reached that this issue was really a matter of a defined objective and how to set priorities rather than an area of concern in the context of items (1), (2), and (3) listed in the preceding paragraph.

Participants generally agreed that the primary objective of fire protection was to prevent the release of radioactive materials to the environment. This objective included safe shutdown of the reactor and protection against release both before and after safe shutdown and during and after a fire, even if safe shutdown is not in question. Protection of personnel and property should be the second priority, and continued plant operation should be third.

Dr. Levenson reported that some disagreement occurred on the relative roles of analysis and inspection in the setting of priorities for upgrades. This disagreement was partially resolved by agreement that no single answer can be found for this question and that the answer varies by type and generation of the plant and even varies from plant to plant of the same type and same generation. Simple inspection is a good starting point for older plants, whereas newer plants that already have many improvements need more sophisticated analysis to ensure that such issues as system interactions and post-fire operations have been adequately addressed.

A subset of the number 1 priority might be (1) any fire whose consequences are the release of radioactivity or the prevention of safe shutdown, (2) any fire that can cause a nuclear accident even if no radioactivity is released, and (3) any fire that could damage those systems needed to mitigate the consequences of an accident.

Very little discussion was heard on the specifics that vary from plant to plant and that are best prioritized by the plant people. Some discussion occurred on the delays in implementation of aid because of the requirements in some cases for testing western components or equipment to Russian or Ukrainian standards.

The workshop did not uncover any overlooked issues or discover any new truths. It probably had two benefits. The first benefit was that the workshop helped provide an independent look at the priorities set by the plant people, and, second, the discussion reminded the participants that even the newest plants would probably benefit from analysis of such concerns as system interactions and post-fire conditions.

In conclusion, Dr. Levenson repeated one of the most significant observations of the first workshop: No improvement in safety occurs unless changes are made at the plant. The changes may be to hardware, software, training, procedures, or personnel, but unless they occur at the plant, safety is not improved.

REMARKS BY IVAN SELIN

Dr. Selin remarked that nuclear safety problems in the Soviet Union and Eastern Europe were related to (1) design inadequacies in some key Soviet-designed plants, particularly the graphite-moderated RBMK; (2) substandard operational safety procedures and attention to detail in managing the production of electricity at all nuclear power plants; and (3) an almost complete lack of independent governmental regulation of the state utilities, which were devoted to fulfilling the production requirements of the latest economic plan, often at the expense of safe operation.

He argued that the nuclear safety problems in these countries are structural and have less to do with engineering and personnel than with economics, sound management, and the difficulty these nations are having in making the transition to market economies. Dr. Selin believes we need to effect an orderly transition in our nuclear safety efforts in Russia and Ukraine, from short-term measures (such as technical fixes, operational improvements, and regulatory practices) to longer-term measures (e.g., assuring adequate resources and making firm institutional and management arrangements).

He further noted that economic stability and market pricing of energy are fundamental to the development and maintenance of a safe and sustainable nuclear power program. Little progress has been made in Russia—and even less in Ukraine—toward the restructuring of their energy economies needed to pay for the safety improvements they so desperately need. Certain conditions must be met for nuclear power to be both economically sound and physically safe. A fundamental, realistic revamping of the energy pricing mechanism and a commitment to provide the maintenance and investment resources needed for technical excellence must occur; only then will significant international investments find their way to the nuclear programs of Russia and Ukraine.

In addition, Dr. Selin argued, sustainable safety requires adoption of certain management principles fundamental to the adoption of a "safety culture." Until Russia and Ukraine themselves institute modern economic reforms, we in the West risk pouring aid resources down a bottomless pit. Truly "sustainable" efforts to improve nuclear safety must emerge from within these societies themselves. These efforts are not yet happening. Plant operators in Russia and Ukraine are not being paid on time, if at all. Utilities are not receiving payment for the electricity they produce. Regulators face overwhelming bureaucratic and legal barriers as they try to do their jobs properly.

Dr. Selin also noted that the second important element of sustainable safety is the establishment of independent, enforceable nuclear regulatory regimes. He argued that three elements of sound regulation are especially important in establishing and maintaining a proper nuclear safety culture.

- First, every nuclear nation must provide a firm legal foundation for a strong and independent regulatory authority to monitor and enforce high levels of safety.
- Second, the regulatory authority must have the resources, in terms of personnel and technical capacity, to be effective.
- Third, both the industry and regulators must apply rigorous and binding standards that cover all safety-relevant aspects of the nuclear fuel cycle.

He pointed out that these are the main elements of the International Nuclear Safety Convention that was opened for signature this past September in Vienna, Austria. The Convention requires each contracting party to take the needed legislative, regulatory, and administrative measures to implement its obligations under the Convention.

Furthermore, Dr. Selin noted that national law or binding international commitment must require a state to put into place legal liability and financial protection arrangements that would provide adequate compensation for damage in the event of a nuclear accident while setting appropriate limits on third-party liability. Such protection holds both the nation and the nuclear power plant operators accountable for protecting the public health and safety while assuring the public every right to redress any injury it might suffer as a result of negligence or improper operation.

Dr. Selin concluded by saying that "nuclear power is not for everyone." Without adequate economic resources, without an energy market where prices for nuclear power are competitive with other forms of electricity production, and without vigorous government regulation, civilian nuclear power becomes dangerous: a technology that, if mismanaged, can contribute to drastic instabilities.

Nuclear units coming on line now can be expected to operate over at least the next 40 years. Nuclear safety cannot be a temporary undertaking that depends on the support of outsiders. Each nation choosing to use nuclear energy to generate electricity must be prepared to make a long-term commitment to establish and maintain the key elements of a nuclear safety culture that will protect the public and the environment over the full fuel life cycle.

CONCLUSIONS

The *Second ANS Workshop on the Safety of Soviet-Designed Nuclear Power Plants* provided an outstanding opportunity for nuclear safety experts with an interest in Soviet-designed power reactors to directly exchange important technical information. The larger-than-expected attendance attested to the strong interest in this subject.

The workshop clearly found that much progress was being made in enhancing the safety operation of these reactors. Although institutional issues such as liability presented an obstacle to faster and wider implementation of safety improvements, many organizations in Western countries—both governmental and private—are nevertheless earnestly moving forward with their Eastern counterparts in the quest for safer operation. Social, economic, and political factors still remain as issues to be reckoned with as nations go forth to improve safety.

A subsequent workshop will be useful to measure and encourage continued safety improvement and to enhance direct communication among nuclear safety professionals in the many countries with a strong interest in improving the safety of Soviet-designed nuclear power plants.

ACKNOWLEDGMENTS

This work was performed under the auspices of the U.S. Department of Energy under contract No. DE-AC02-76CH00016. The Nuclear Reactor Safety Division of ANS was the prime motivator and sponsor from within the Society. Members of the U.S. Nuclear Regulatory Commission, Department of State, and the International Atomic Energy Agency also cooperated in the organization of this meeting and made essential contributions to its structure and content.

Finally, many people at Brookhaven National Laboratory provided outstanding support that led to the success of the workshop.

REFERENCE

1. R. A. Bari (Ed.), *Second ANS Workshop on the Safety of Soviet-Designed Nuclear Power Plants, Summary Report*, Report BNL-52457 (CONF-941196-Summ.), Brookhaven National Laboratory, March 1995.

Table 1 Technical Program of the Second ANS Workshop on the Safety of Soviet-Designed Nuclear Power Plants[a]

OPENING SESSION

Opening Remarks R. A. Bari, *Workshop Organizer*
ANS Welcome A. Walter, *ANS President*
Keynote Address T. Lash, *Director, Office of Nuclear Energy, U.S. Department of Energy*

PLENARY SESSION I—"Government and Institutional View"

Co-Chairs: R. Budnitz, *FRA, USA*
D. Giessing, *DOE, USA*
Panelists: N. Steinberg, *SCNRS, Ukraine*
Z. Szonyi, *AEC, Hungary*
E. Ignatenko, *Rosenergoatom, Russia*

PLENARY SESSION I—"Safety Enhancements Status Reports"

Chairperson: E. Warman, *Stone and Webster, USA*
Panelists: M. Iankov, *Kozloduy, Bulgaria*
J. Vokurek, *CEZ, Czech Republic*
V. Bronnikov, *Zaporizhzhya, Ukraine*
J. Suchomel, *NPPRI, Slovakia*

PLENARY SESSION III—"Challenges for Commercial Sector in Long Term Efforts to Improve Nuclear Safety"

Co-Chairs: J. Dobrin, *Department of State, USA*
N. Ponamarev-Stepnoi, *Kurchatov, Russia*
Panelists: H. Shapar, *Shaw, Pittman, Potts, and Trowbridge, USA*
F. Demarcq, *EBRD*
J. Harper, *OPIC, USA*
R. Bennet, *Westinghouse, USA*

RECEPTION AND BANQUET

Introduction: R. B. Duffey, *BNL, USA*
Speaker: I. Selin, Chairman, *U.S. NRC*

WORKING SESSIONS

1) RBMK

Chairperson: J. Vilemas, *LEI, Lithuania*
Facilitator/Rapporteur: M. Hayns, *AEA Technology, UK*

2) VVER 230

Co-Chairs: J. Suchomel, *NPPRI, Slovakia*
V. Evanov, *Rosenergoatom, Russia*
Facilitator/Rapporteur: A. Birkhofer, *GRS, Germany*

3) VVER 213

Co-Chairs: P. Trampus, *PAKS, Hungary*
F. Pazdera, *NRI Czech Republic*
Facilitator: W. R. Johnson, *University of Virginia, USA*
Rapporteur: M. Fontana, *ORNL, USA*

4) VVER 1000

Co-Chairs: M. Komsi, *IVO, Finland*
S. Krylov, *Rosenergoatom, Russia*
Facilitator/Rapporteur: W. Horak, *BNL, USA*

5) Regulation. Standards. Criteria

Co-Chairs: J. Stuller, *State Office for Nuclear Safety, Czech Republic*
A. Bukrinsky, *Gosatomnadzor, Russia*
Facilitator/Rapporteur: A. Carnino, *IAEA, Austria*

6) Experiences of Western Contractors

Co-Chairs: P. Yanev, *EQE, US*
J. P. Baret, *EdF, France*
Facilitator/Rapporteur: D. Squarer, *USA*

7) Training. Procedures. Operational Safety

Co-Chairs: A. Kroshilin, *VNIIAES, Russia*
W. Hastie, *WANO, USA*
Facilitator/Rapporteur: C. Heising, *Iowa State University, USA*

8) PRA Results and Applications to Safety Improvement

Co-Chairs: L. Voross, *IEPR, Hungary*
E. Soderman, *Energy and Safety, Sweden*
Facilitator/Rapporteur: J. Bickel, *INEL, USA*

Opening Remarks: G. Vajda, *AEC, Hungary*
"The AGNES Projects"

9) Containment/Confinement Enhancements

Co-Chairs: E. Koshmanov, *Atomenergo Project, Russia*
J. Misak, *NRA, Slovakia*
Facilitator/Rapporteur: W. Deitrich, *ANL, USA*

10) Fire Protection/Fire Fighting

Co-Chairs: E. Ivanov, *Rosenergoatom, Russia*
E. Diatian, *ENERGOPROEKT, Ukraine*
Facilitator/Rapporteur: M. Levenson, *USA*

WORKSHOP LUNCHEON

Introduction: A. D. Hink, *AECL, Canada*
Speaker: H. Kopchynsky, *NUCON, Ukraine*
"Status of Chornobyl"

SUMMARY SESSION

Co-Chairs: L. Dodd, *PNL, USA*
T. Speis, *U.S. NRC*

- Summary of Plenary Sessions
  A. D. Rossin, *Rossin & Associates, USA*
- Summaries of Working Sessions by Rapporteurs
- Closing Discussion

(Table continues on the next page.)

Table 1 (Continued)

[a]Abbreviations of organizations:

AEA Technology	Atomic Energy Authority Technology	IVO	Imatram Voima
AEC	Atomic Energy Commission	LEI	Lithuanian Energy Institute
AECL	Atomic Energy of Canada Ltd.	NPPRI	Nuclear Power Plants Research Institute
ANL	Argonne National Laboratory	NRA	Nuclear Regulatory Authority
BNL	Brookhaven National Laboratory	NRI	Nuclear Research Institute
CEZ	Czech Power Enterprises, Inc.	NRC	Nuclear Regulatory Commission
DOE	Department of Energy	NUCON	Nuclear Consultant
EBRD	European Bank for Reconstruction and Development	OPIC	Overseas Private Investment Corp.
EdF	Electricite de France	ORNL	Oak Ridge National Laboratory
FRA	Future Resources Associates, Inc.	PNL	Pacific Northwest Laboratories
GRS	Gesellschaft für Reaktorsicherheit	SCNRS	State Committee for Nuclear and Radiation Safety
IAEA	International Atomic Energy Commission	VNIIAES	All-Union Scientific Research Institute of Atomic Power Plants
IEPR	Institute for Electric Power Research		
INEL	Idaho National Engineering Laboratory	WANO	World Association of Nuclear Operators

Elements of a Nuclear Criticality Safety Program

By Calvin M. Hopper[a]

Abstract: *Nuclear criticality safety programs throughout the United States are quite successful, as compared with other safety disciplines, at protecting life and property, especially when regarded as a developing safety function with no historical perspective for the cause and effect of process nuclear criticality accidents before 1943. The programs evolved through self-imposed and regulatory-imposed incentives. They are the products of conscientious individuals, supportive corporations, obliged regulators, and intervenors (political, public, and private). The maturing of nuclear criticality safety programs throughout the United States has been spasmodic, with stability provided by the volunteer standards efforts within the American Nuclear Society. This presentation provides the status, relative to current needs, for nuclear criticality safety program elements that address organization of and assignments for nuclear criticality safety program responsibilities; personnel qualifications; and analytical capabilities for the technical definition of critical, subcritical, safety and operating limits, and program quality assurance.*

As the title indicates, this presentation provides **a** description of the elements of **a** nuclear criticality safety program (notice the omission of the word, **the**). In some instances it offers information that is rather ordinary; in other instances it is moderately to substantially controversial. In almost all cases the information and concepts of what constitutes the elements of a nuclear criticality safety (NCS) program are drawn from the resources of many people and many organizations. All the elements have some relevance to regulatory oversight programs as well as to regulated and applied programs at facilities that process, store, or transport fissionable materials aside from reactors. The elements identified for this presentation are (1) organization of and assignments for NCS program responsibilities, (2) personnel qualifications, and (3) tools for evaluating subcriticality to establish safety and operating limits.

It is difficult to imagine that anything new could be offered in the way of describing the elements of a nuclear criticality safety program; however, occasional introspection from a removed perspective with a different emphasis or current thought can be interesting if not useful. This article evolved for the purpose of introspection—a little history, some sage advice from people who have led or are leading the nuclear criticality

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safety community, a reminder of recent experiences, current events, and value systems that are shaping the safety business.

The definition of nuclear criticality safety is recognized in our industry's standard ANSI/ANS-8.1, *American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*, as "protection against the consequences of an inadvertent nuclear chain reaction, preferably by prevention of the reaction."[1] The bases of this definition, in terms of consequences, however, and the reason for the administration of nuclear criticality safety will not be allowed to escape this presentation (i.e., to paraphrase the introduction to Ref. 1: The protection of life is more important than the protection of property).[2] Of course, in many instances the protection of property is what permits the protection of life.

NCS PROGRAM ELEMENTS

In 1973, Roy Reider, an "old salt" in the business of overall safety who was the safety director of Los Alamos Scientific Laboratory, shared his thoughts[3] on the fundamental elements of safety. As applied to safety in general, he identified **the most important** element of safety as the assignment and acceptance of responsibility. He went on to identify other elements, such as management leadership in the declaration of policy and assumption of responsibility for control of accidents; the establishment of requirements for procedures, including review of procedures; maintenance of safe working conditions, including inspections by specialists (of cranes, elevators, high-pressure equipment, fire protective devices, etc.), committee inspections, proper purchasing and acquisition, and supervisory interest; safety training for supervisors and employees, which could include first aid, emergencies, review of accidents, technical information, protective clothing, safety fundamentals, and a variety of specific subjects; medical and first aid; and a system for reporting and recording accidents, including near misses or potential mishaps, which can alert personnel to needed protective measures or procedural changes.[a]

[a]To Mr. Reider, all of the above "other elements" are necessary but not sufficient without a sense of personal urgency in the attention to safety by people responsible for or performing work. This includes management's responsibility in the assignment of work and facilitating resources as well as the worker's and supervisor's acceptance of the assignment and the concomitant responsibilities.

It should come as no surprise that those fundamentals deviate only so very slightly from the guidance found in our ANSI/ANS-8.1, *American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors,* and ANSI/ANS-8.19, *American National Standard Administrative Practices for Nuclear Criticality Safety.*[4] Adoption of guidance from all the ANS-8 standards is voluntary. According to the introductory "boiler plate" statements in the standards:[1]

> An American National Standard implies a consensus of those substantially concerned with its scope and provisions. An American National Standard is intended as a guide to aid the manufacturer, the consumer, and the general public. The existence of an American National Standard does not in any respect preclude anyone, whether he has approved the standard or not, from manufacturing, marketing, purchasing, or using products, processes, or procedures not conforming to the standard.

All of the fundamentals that are provided within the standards are subject to interpretation, definition, application, improvisation, administration, and regulation. This means that all of these "-tions" are prone to revision and accentuation on the basis of perceived needs—perceived by nuclear criticality safety specialists, safety managers, safety review committees, operating personnel and supervision, site (business) managers, regulators, legislators, and the public. Obviously, these perceived needs then become a motivating force, at different levels of control, for shaping the depth, breadth, and quality of our nuclear criticality safety programs into what they are becoming today.

Because the voluntarily developed standards provide **general** consensus requirements (the shalls) and **general** consensus recommendations (the shoulds), the user must fill in the details of how to implement the guidance. Increasingly, the adequacy or quality of the user-defined details is judged or even specified by the regulator.

The following is a discussion of the elements of a nuclear criticality safety program and how they appear to be evolving.

Organization of and Assignments for NCS Program Responsibilities

As previously stated, **the most important** fundamental element of safety is the assignment and acceptance of responsibilities by operating officials, safety and health personnel, supervisors, and technical committees. In ANSI/ANS-8.1, the requirements for establishing responsibility are distinct but general: "Management shall clearly establish responsibility for nuclear criticality safety." In ANSI/ANS-8.19, it is stated differently: "Management shall accept overall responsibility for safety of operations." These statements provide the stage and plot for a comprehensive nuclear criticality safety program, but they do not specify the details of the scenery for the stage. The ANS-8 standards were developed to be "what to do" standards as opposed to "how to do it" standards. Even the following more particular requirements from ANSI/ANS-8.19 are not intended to specify the details of how to accomplish the more general requirements of ANSI/ANS-8.1.

Management Responsibilities

Fulfilling the requirement of accepting and assigning responsibility[a] for nuclear criticality safety necessarily involves the development of details that perform as follows:

- Formulate nuclear criticality safety policy and make it known to all employees involved in operations with fissile material.
- Assign responsibility and delegate commensurate authority to implement the established policy.
- Provide personnel familiar with the physics of nuclear criticality and with associated safety practices to furnish technical guidance appropriate to the scope of operations.
- Establish the criteria to be satisfied by nuclear criticality safety controls.
- Establish a means for monitoring the nuclear criticality safety program.
- Ensure that before a new operation with fissionable materials is begun or before an existing operation is changed, it shall be determined that the entire process will be subcritical under both normal and credible abnormal conditions.
- Ensure management participation in auditing the overall effectiveness of the nuclear criticality safety program.

The standards recommend that supervision at all levels of management should be made as responsible for nuclear criticality safety as for any of their other functions. Although this is a recommendation, we are seeing it applied more frequently as a measure of job performance at all levels of employment. The standards acknowledge that nuclear criticality safety differs in no intrinsic way from industrial safety, and good managerial practices apply to both. This is to say that it should be assigned in a manner compatible with that for other safety disciplines and should, to the extent practicable, be administratively independent of operations. This is also a recommendation, but it is becoming common throughout the industry to observe nuclear criticality safety organizations reporting to a first- or second-level manager under the site manager.

Supervisory Responsibilities

Accompanying management's responsibilities are the responsibilities of supervision. Supervision shall:

- Accept responsibility for the safety of operations under their control.
- Be knowledgeable in those aspects of nuclear criticality safety relevant to operations under their control.
- Provide documented training and verification of understanding by personnel under their supervision and ensure that they have an understanding of procedures and safety considerations so they may perform their functions without undue risk.
- Participate in the development and maintenance of written procedures that include those controls and limits significant to nuclear criticality safety and that facilitate the safe and efficient conduct of the operation.
- Review active procedures periodically.
- Ensure that new or revised procedures impacting nuclear criticality safety are reviewed by the nuclear criticality safety staff.
- Verify compliance with nuclear criticality safety specifications for new or modified equipment before its use.
- Require conformance of operations with good safety practices, including unambiguous identification of fissile materials and good housekeeping.
- Control the movement of fissile materials.
- Provide appropriate material labeling and area postings specifying material identification and all limits on parameters that are subject to procedural control.
- Control access to areas where fissile material is handled, processed, or stored.
- Exercise control for the continued presence and intended distributions and concentrations of neutron-absorbing materials used to ensure subcriticality.

[a]The responsibilities must be assigned to and accepted by qualified and competent individuals to support an effective nuclear criticality safety program.

• Provide for the detection of deviations from operating procedures and unforeseen alterations in process conditions that affect nuclear criticality safety and ensure that they are documented, reported to management, investigated promptly, and prompt actions taken to prevent recurrence.

• Prepare emergency procedures that are approved by management, clearly designate evacuation routes and personnel assembly stations to be used for annual drills and training, ensure that injured or exposed personnel receive proper care and treatment, provide for necessary instrumentation for assessing radiation fields and for immediate identification of exposed individuals, and ensure that appropriate installation of nuclear criticality accident alarms is accomplished.

• Review operations at least annually to ensure that procedures are being followed and that process conditions have not been altered so as to affect the nuclear criticality safety evaluation.

• Ensure that before a new operation with fissionable materials is begun or before an existing operation is changed the entire process will have been determined to be subcritical under both normal and credible abnormal conditions.

Many of the preceding requirements are interwoven with the responsibilities of management to provide commensurate authority and appropriate resources to address the responsibilities. Obviously, there are additional overlaps of responsibility with the nuclear criticality staff to provide analytic and supportive resources to supervision and management.

The preceding supervisory responsibilities are not limited to production supervision only. Clearly, maintenance activities on ancillary equipment, perhaps removed from the fissile material operating area, could have adverse impacts on nuclear criticality safety and require maintenance supervision training, awareness, and positive controls to maintain a sound nuclear criticality safety program. Where professional trainers are employed for the development of training programs, there must be communication and concurrence among operating supervision and nuclear criticality safety staff about detailed training program content. Similarly, safeguards personnel have influential functions in nuclear criticality safety. Personnel accountable for nuclear material know the location and status of fissile material in near real time. Security forces have important missions in response to emergency situations. Other personnel, such as fire fighters and emergency medical staff, also have the potential to influence nuclear criticality safety by entering fissile material processing or storage areas and disrupting approved configurations. Analytical laboratory personnel have key functions in the identification of fissile material forms and concentrations having direct and indirect impact on nuclear criticality safety. The engineering and design organizations at facilities, acting in concert with operations personnel and the nuclear criticality safety staff, have an opportunity to make significant contributions to safety through design.

Nuclear Criticality Safety Staff Responsibilities

Because operational line management and supervision have production responsibilities and are equally responsible for nuclear criticality safety, production, development, and research, as well as other functions, they need the cooperation of the nuclear criticality safety staff to furnish technical guidance appropriate to the scope of operations. The nuclear criticality safety staff, to the extent practicable, should be administratively independent of operations. Although management may use consultants and nuclear criticality safety committees to achieve the objectives of the nuclear criticality safety program, the more desirable alternative is to have resident personnel who are familiar with the operations and can help maintain a corporate safety memory for the facility. The responsibilities of the nuclear criticality safety staff include the following:

• Provide technical guidance for the design of equipment and processes and for the development of operating procedures.

• Maintain familiarity with current developments in nuclear criticality safety standards, guides, and codes.

• Maintain familiarity with all operations within the organization requiring nuclear criticality safety controls.

• Assist supervision, on request, in training personnel.

• Participate in or conduct audits of criticality safety practices and compliance with procedures as directed by management.

• Examine reports of procedural violations and other deficiencies for possible improvement of safety practices and procedural requirements and report examination findings to management.

• Review new or revised operating procedures impacting nuclear criticality safety.

• Provide expertise in the performance of clear, detailed, and confirmed process evaluations (explicitly identifying controlled parameters and their limits) that determine the subcriticality of the entire process under both normal and credible abnormal conditions.

- Provide offered or requested assistance to management or supervision in the fulfillment of their responsibilities, such as participation or leadership in the development of emergency plans, in the development of personnel training materials, in operational supervisory audits, and in the interpretation of regulatory requirements.

Other ANS-8 standards exist that provide more specific guidance. It is reasonable to expect the nuclear criticality safety staff to assist line management in their interpretation and implementation.

ANSI/ANS-8.3,[5] *Criticality Accident Alarm System*, provides the performance criteria for detecting nuclear criticality accidents. Although instrumentation, health physics, or emergency response personnel might refer to this standard directly, the nuclear criticality safety staff could assist these personnel with understanding the hazard and interpreting the standard for application.

ANSI/ANS-8.5,[6] *Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material*, describes the chemical and physical environment for usage, properties of the rings and packed vessels, maintenance inspection procedures, and criticality operating limits for solution systems containing ^{235}U, ^{239}Pu, or ^{233}U. Although operating line management and supervision may use the standard for the application of Raschig rings in their operations, the nuclear criticality safety staff plays a central role in the approval for use and quality assurance of the application.

The use of ANSI/ANS-8.7,[7] *Guide for Nuclear Criticality Safety in the Storage of Fissile Materials*, and ANSI/ANS-8.15-1987,[8] *Nuclear Criticality Safety Control of Special Actinide Elements*, is interpreted and applied almost solely by the nuclear criticality safety staff because of the nature of the standards.

ANSI/ANS-8.20,[9] *Nuclear Criticality Safety Training*, provides criteria for the administration of a nuclear criticality safety training program for personnel who manage or work in or near facilities where potential exists for a criticality accident outside reactors. It does not apply to the training of nuclear criticality safety staff. Although this standard is directly usable by line supervision or a training organization, quality implementation uses the technical expertise of nuclear criticality safety staff.

It is apparent that the organization of and assignments for nuclear criticality safety program responsibilities are as complicated and compounded as any other program to prevent or mitigate hazardous, sensitive, or expensive accidents. The details of satisfying regulatory oversight have been fairly muddled to date; however, the regulators are trying to address these details through the development of guides for the performance of government contractor nuclear criticality safety programs and licensing review plans that describe items for review, review procedures, and acceptance criteria.

Personnel Qualifications

The industry began with highly motivated individuals, with exceptional academic credentials, working in collegial environments where the safety of the project and the workers was a well-formulated commitment; however, the individual was expected to and did take personal responsibility for his own and his co-workers' safety in attics, log cabins, warehouses, production lines, and towering superstructures.[10] Today the regulatory expectations extend beyond the poorly documented practical training of those early years. Today's workers and nuclear criticality safety staff are expected to be formally trained and qualified. Also, the demonstration of their training, the relevance of the training to their job, and their qualifications to perform the job are compared with prescribed acceptance criteria and documented. These requirements have become formidable.

The evolution of rigorous personnel qualification requirements was historically driven by the high degree of sensitivity to the potential release of terrible quantities of fission products from power reactors.[11] It is not unreasonable to be sensitive to the possibility of uncontrolled release of fission products from power reactors [for example, the accident at the 3200-MW(t) Chernobyl Nuclear Power Station, Unit 4[12]]. These assumptions can be made: (1) about 7×10^{16} fissions per pound of H.E.[13] energy, (2) an equivalency of H.E. and TNT, (3) an atomic bomb explosive yield of 12.5 kilotons (2.5×10^7 pounds) of TNT,[10] and (4) 8.6×10^{24} fissions per day[14] for a 3200-MW(t) power reactor. One could say that a Chernobyl power reactor is producing, in a controlled way, the fission products of 4.9 atomic bomb explosions on a daily basis. These inflammatory mathematics are not relevant to the relative hazards between an atom bomb and controlled nuclear fission, nor are they relevant to the nuclear facility criticality accidents that nuclear criticality safety staffs seek to prevent (between about 10^{17} and 10^{19} fissions[2]). There is, however, a perceived need for regulatory prescriptions to provide similar standards of rigorous, high-quality, documented nuclear criticality safety training for all nonreactor nuclear facility workers that may influence the safety of fissionable materials (i.e., laborers, craftsmen, technicians, foremen, supervisors, line managers, facility managers, and technical support personnel—nuclear criticality safety staff).

One of the most recent developments in regulatory prescriptions is DOE Order 5480.20[15] on reactor and nonreactor nuclear facility personnel selection, qualification, training, and staffing requirements. Order 5480.20 is a spin-off of various related NRC Regulatory Guides, American National Standards, the Institute of Nuclear Power Operation's performance-based training accreditation program, and Title 10 *Code of Federal Regulations* Part 55 on the training of senior reactor operators and reactor operators. The following discussion is offered as a glimpse of the regulatory future and provides a moderately to substantially controversial subject. Although the particular referenced DOE Order 5480.20 addresses selection, qualification, training, and staffing of all reactor and nonreactor nuclear facility personnel, only the issue of nonreactor nuclear facility nuclear criticality safety technical staff will be addressed.

DOE Order 5480.20 has a main body of text with subparagraphs addressing (1) purpose, (2) cancellation of other Orders relevant to the subject, (3) scope, (4) references, (5) background of development, (6) definitions, (7) to whom the Order is applicable, (8) responsibilities of DOE personnel and organizations, (9) program requirements, and (10) implementation schedule. The main body of text is followed by four chapters entitled (I) General Requirements, (II) Category A Reactor Personnel, (III) Category B Reactor Personnel, and (IV) Nonreactor Nuclear Facility Personnel. Chapter I provides general requirements for a training program, a training organization, subcontractor personnel qualifications, personnel selection, the qualification and certification processes, training, operator and supervisor examination and reexamination, requalification, exceptions to training, extension, alternatives to education, alternatives to experience, limitations for overtime worked, and records requirements.

The following is a reasonable interpretation of what the education and experience qualifications must be for a new, developing, or transitioning nuclear criticality safety staff, with two exceptions:

- "Interpretive guidance" has been issued by DOE to modify the words or meanings of the words in Chapter II, subparagraph 2.c. and Attachment II-1, and in Chapter IV, subparagraph 2.f.
- The category of nuclear criticality safety specialist has been omitted from contractor-submitted and DOE-approved "Training Implementation Matrix."

The definition of and entry-level requirements for technical support staff are provided in subparagraph 2.f. of Chapter IV as:

> Personnel in these positions are responsible for supervision and performance of technical support functions for the operating organization. Personnel involved in surveillance, testing, analyzing plant data, planning modifications, program review, and technical problem resolution in their area of expertise are also included. They have expertise in mechanical, electrical, instrumentation and control, chemistry, radiation protection, training, safety, or quality assurance. For personnel assigned to equivalent positions, non-reactor nuclear facilities should use the education and experience requirements contained in Chapter II, Category A Reactor Personnel, subparagraph 2.c.

The subparagraph provides education and experience requirements for which there are no equivalent reactor and non-reactor positions.

Chapter II, subparagraph 2.c. defines technical support personnel and provides basic education and experience requirements for such personnel as the following:

> Personnel in these positions are responsible for supervision and **performance** of technical support functions for the operating organization. Personnel involved in **surveillance, testing, analyzing plant data, planning modifications, program review, and technical problem resolution in their area of expertise** are also included. They have expertise in mechanical, electrical, instrumentation and control, chemistry, radiation protection, **training, safety, quality assurance,** or **reactor engineering**. Unless otherwise stated, the basic education requirement is a baccalaureate in engineering or related science; the experience requirement is 2 years job-related, of which 1 year shall be nuclear experience. Education and experience requirements are intended to apply to supervisory positions or positions with **authority to review and concur,** and not to entry-level positions.

Subparagraphs 2.c.(1–9) identify nine different job positions in the category of technical support personnel: (1) reactor engineering, (2) instrumentation and control, (3) chemistry and radiochemistry, (4) radiation protection, (5) preoperational testing engineer, (6) startup testing engineer, (7) training coordinator, (8) training instructor, and (9) shift technical advisor. A nuclear criticality safety specialist position is not specifically identified.

Typically, a nuclear criticality safety staff specialist has a technical support safety role for:

- Analyzing plant data
- Planning modifications

- Program review
- Technical problem resolution in their area of expertise (nuclear criticality safety)
- Nuclear criticality safety
- Quality assurance of fabricated equipment and process procedures
- Subcritical reactor engineering

Additionally, unless in the role of a trainee, a nuclear criticality safety specialist is obligated to exercise **authority to execute, review, and concur** in the performance or review of nuclear criticality safety evaluations, safety analyses, and operational approvals for facility equipment and processes.

The most nearly equivalent of the nine identified positions is the reactor engineering position because of the special requirement for knowledge and experience of reactor physics. Also, every nuclear criticality safety evaluation, safety analysis, and fissionable material process requires subcritical reactor engineering design work. The educational and experience qualifications for (subcritical) reactor engineering positions of review and concurrence are a baccalaureate in engineering or related science, 4 years of job-related experience that shall include 2 years of nuclear experience and 6 months on site. For simplistic reference purposes, radiation protection personnel with similar review and concurrence responsibilities must have a baccalaureate in engineering or related science, 4 years of job-related experience that shall include 3 years of nuclear experience and 6 months on site.

Chapter I provides general training requirements for certain positions that include nuclear criticality safety specialists. For instance,

- General Employee Training (GET) that includes
  - General description of facilities
  - Job-related policies, procedures, and instructions
  - Radiological health and safety program
  - Facility emergency plans
  - Industrial safety/hygiene program
  - Fire protection program
  - Security program
  - Quality assurance program
- Probabilistic risk assessment (PRA) training in facilities for which a PRA has been performed
- Technical support personnel training in facility-specific subject areas pertinent to their areas of responsibility that include
  - Facility organization
  - Facility fundamentals
  - Facility systems, components, and operations
  - Environment, safety, and health orders
  - Codes and standards overview
  - Facility document system
  - Safety analysis reports and technical specifications/operational safety requirements
  - Nuclear criticality control
  - Material, maintenance, and modification control
  - ALARA and radwaste reduction program
  - Quality assurance and quality control practices
  - Performance-based training in their area of responsibilities

Requalification training, testing, and certification are to occur biennially. Exceptions from such training requirements shall be approved by contractor management in accordance with contractor-developed and DOE-approved training exception plans.

The preceding qualifications and requirements for the training of a nuclear criticality safety specialist are substantial on the basis of the preceding interpretation. Furthermore, details and burdens of additional required performance-based training for the development and qualification of a nuclear criticality safety specialist are likely to be controversial; however, such a description was proposed for discussion at the American Nuclear Society meeting in San Francisco.[16] The purpose of the description was to define a training and qualification program that will meet the intent of the 5480.20 performance-based training concept while recognizing distinctions of particular job assignments. It recognizes the possibility of independent functional specialties (e.g., regulatory, computational, training development/execution, auditing, operations, and processing); modes of training (e.g., formal on- and off-site, apprenticeship, and professional development activities); and classification of job progression (e.g., entry level, apprentice, specialist, senior or lead specialist).

Although the preceding training requirements, interpreted from DOE Order 5480.20, will likely produce thoroughly trained personnel, it is not clear that all the training rigor and documentation is justifiable from a practical viewpoint given academic preparation and trainee on-the-job development of expertise. This is even more questionable when compared with the preparation and qualification of persons assigned responsibility for evaluating and controlling far more serious and frequently occurring industrial and health hazards. Irrespective of philosophies underlying the issue of training program content and quality, nonreactor nuclear facility training regulatory requirements are developing as described.

Tools for Evaluating Subcriticality to Establish Safety and Operating Limits

Apart from accepting the assignment for nuclear criticality safety responsibilities as a resource to line management and support organizations and aside from accommodating the concomitant training and qualification or certification requirements, most nuclear criticality safety specialists must have the tools of the trade available to them so that they can actually perform a function in support of nuclear criticality safety. Tools consist of the following:

- Published and referable experimental data reports (critical and subcritical integral and differential experimental measurements and differential cross-section measurements).
- Computational methodologies and computer programs.
- Technical journals and documents reporting results of computer code verification and validations and computational studies revealing characteristics of various fissionable material systems that may be pertinent to on-site evaluations and analyses.
- Industrial and nuclear criticality safety "near miss" and accident evaluation and analysis reports.
- Regulatory documents relevant to nuclear criticality safety requirements.
- Industry consensus standards on subjects applicable to nuclear criticality safety, such as facility equipment and human reliability data and analysis techniques.

Viable nuclear criticality safety organization staffs maintain an active relationship with professional organizations such as the Nuclear Criticality Safety and the Mathematics & Computations Divisions of the American Nuclear Society that provide a focus on nuclear criticality safety issues and relevant computational methodologies. Such relationships provide a window to information resources through journals, publications, and technical meetings.

Because of the limited database of critical and subcritical experimental measurements relative to actual facility equipment or process evaluations, a nuclear criticality safety staff is required to rely on quality-assured computational evaluations to demonstrate subcriticality and to develop safety and operating limits. Computer codes (software) used to evaluate subcriticality of fissionable material configurations considered for establishment of safety and operating limits are required to be verified and validated.[1] To remain so through a computational evaluation period, the software must be configuration controlled to ensure consistency in operation. Emphasis to use ANSI standards is increasing to ensure quality assurance of software requirements that will help establish safety and operating limits within nuclear facilities.[17] These quality requirements involve the development of quality assurance plans[18] for software verification and validation[19] and configuration control[20] to be used by authorized qualified personnel.

The use of verified and validated quality-assured software by qualified personnel is generally recognized as a prerequisite to performing nuclear criticality safety evaluations. The validation process is assumed to provide for the validation of neutron cross sections and permit the identification of computational biases. This assumption is "probably" a first-order "truth" for fissionable material systems having similar (whatever that is!) materials and neutron energy spectrum and system geometries as the validation "benchmark." This assumption is "probably not" a first-order "truth" for fissionable material systems dissimilar from the validation "benchmark" by having different materials or different neutron energy spectra dependent on neutron energy utilizations (e.g., n-f, n-n, n-γ, n-2n, n-p, and other reactions for neutron interaction, leakage, and return) and having different physical geometries and interactions that enhance the neutronic differences. Trying to determine the significance of such differences and the assignment of computational biases complicates the nuclear criticality safety specialist's life, especially when challenged for statistically assured quantitative definitions of margins of subcriticality. Consider what we may know about the reactivity effects of the $S(\alpha,\beta)$ thermal scattering treatment on a high neutron leakage system (about 50% leakage) of liquid tributyl phosphate $[(C_4H_9)_3PO_4]$ at different temperatures up to 250 °C as contaminated with various concentrations of 5 wt % enriched uranium. How common is the knowledge that it doesn't matter? How common is the knowledge that it does matter? Is either position defensible with documented information? Without more information and data, I'm in trouble for quantifying any answer. Although this is perhaps an extreme example, on occasion the specialist is confronted with stretching the boundaries of the ill-defined, and probably justified, "area of applicability" for the use of a calculational method.

The preceding is a sticky situation for which there is no handy, definite answer except hypothesizing a perceived conservative model. The business of proper computational evaluations is an element of nuclear criticality safety that is equally important to (1) the description of the process to be evaluated; (2) the description of all contingent conditions having a potential adverse effect on nuclear criticality safety of the operation; (3) the selection of highly reliable controls to prevent the occurrence of considered contingent conditions; and (4) the provision of unmistakable instructions, limits, postings, and training of persons performing the desired operation.

SUMMARY

Although all the elements discussed are exceedingly important to the application of nuclear criticality safety, they are trivial in contrast to **the most important** primary element of safety, which is the assignment and acceptance of responsibility for nuclear criticality safety by each employee having immediate or potential influence on fissionable material operations safety. This, of course, means that every avenue of accident prevention must be exhausted before we have fulfilled our responsibility for nuclear criticality accident prevention. **It is the nuclear criticality safety specialist's responsibility to remain focused on the real issue: The protection of life is more important than the protection of property.**

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Rickover, Excellence, and Criticality Safety Programs

By R. E. Wilson[a]

Abstract: *In a 1983 analysis of the accident at Three Mile Island, Admiral Hyman Rickover, father of the nuclear navy, laid out seven criteria for competence in managing nuclear programs: (1) a rising standard of adequacy, (2) technical self-sufficiency, (3) ability to face facts, (4) respect for radiation, (5) recognition of the importance of training, (6) a concept of total responsibility, and (7) the capacity to learn from experience. These principles remain valid and relevant for today's nuclear industry in general and are applied here to criticality safety programs in particular.*

In January 1954 a historic event occurred in the field of nuclear science and engineering: the launching of the first nuclear submarine. This event inaugurated the productive engineering use of nuclear energy. Credit is rightly laid at the feet of Admiral Hyman Rickover. Rickover demonstrated that nuclear energy could be applied to practical problems like propulsion and could be safely managed.

Many of his associates admired, and perhaps worshiped, the Admiral for his accomplishments. Within the nuclear navy, his system was highly successful. One of his biographers notes, however, that his qualities included an interactive style that was invective and destructive as well as frequently unfair.[1] In conversation he was said to be demanding in the extreme. Perhaps as a result, the considerable influence he exerted on the nuclear world outside his control was a mixed blessing. When some of his associates took important positions elsewhere, they demonstrated that they had absorbed his negative personal style more effectively than his considerable wisdom. In reaction, many segments of the nuclear industry have not yet gained full advantage from the experience of the nuclear navy.

In March 1979 another but darker historic event occurred in the nuclear industry: the accident at Three Mile Island Nuclear Station Unit-2 (TMI-2). The TMI-2 accident was a watershed event with public relations consequences that in many parts of the world exceeded even so serious an accident as the one at Chornobyl.

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The self-confidence of the industry and the public's confidence in industry's competence to manage the nuclear enterprise were both drastically changed. In the aftermath, General Public Utilities (GPU) had an undamaged reactor, TMI-1, at the same site sitting idle.

One nuclear enterprise, however, had retained public confidence. In 1983, GPU asked the architect of this phenomenon to assess the organization and its competence to manage TMI-1. Hyman Rickover agreed, spent the required effort, and issued a report.[2] The report offered a useful evaluation of the utility, but its lasting value lies in its vision. The Admiral distilled the wisdom part of his legacy and laid out seven criteria for competence in managing nuclear programs:

- Rising standard of adequacy
- Technical self-sufficiency
- Facing facts
- Respect for radiation
- Importance of training
- Concept of total responsibility
- Capacity to learn from experience

The GPU wisely made the report public. Any organization responsible for managing a nuclear program would do well to measure itself by Rickover's yardstick. In particular, it would be useful in measuring criticality safety programs.

RISING STANDARD OF ADEQUACY

Rickover's first criterion was that the safety standards of a well-run facility "must be built upon rising standards of excellence which substantially exceed those used for licensing purposes." It is difficult for a staff to maintain enthusiasm for a safety philosophy and standards that are imposed on a facility from the outside. A more serious problem, however, is that, when the safety program is imposed by the government rather than by corporate headquarters, these standards by the nature of the regulatory development process define a minimal base. Meeting these bare requirements results in safety programs that just get by. A competent safety program should strive for an "A" or a "B" grade rather than a "generous C." (Meeting minimal criteria was also called a "gentleman's C" in more chauvinistic days.)

One of the problems of living with a minimal safety program is the low tolerance for error. Programs are not static. They get better or worse, not just older. This truism applies to overall program elements as well as to the individual risk of a unit operation. A facility needs to ensure that the vector of its safety program is up. The alternative will likely prove, at best, embarrassing.

When visiting troubled facilities, one often hears that "we must be safe because we have a license." Sometimes these facilities have a manager of regulatory compliance, but not a manager of safety. The idea may be that, if the government has a safety program, then the facility need not have one. These facilities would fail Rickover's first criterion of competence. He considered an organization living with a subsistence-level safety program to be irresponsible. The Nuclear Regulatory Commission (NRC) investigation of the 1979 TMI-2 accident found a fundamental problem to be that "many utilities apparently regarded bare compliance with NRC regulations as more than adequate for safety."[3]

Each nuclear facility should have criticality safety program goals that represent their own serious effort and of which they are proud. If the effort is competent, compliance with government requirements should not be a serious concern. As goals are reached, they should be reassessed and, if appropriate, strengthened. Regulators should encourage rather than impede this process.

The concept of rising standards of adequacy applies also to the larger field of criticality safety. The premise of professional meetings is that we can do better and need to communicate our studies and our experiences to do so. The larger criticality safety community would do well to document programs that could be considered "A" or "B" grade, however controversial these might be, as a way of assisting individual facilities.

TECHNICAL SELF-SUFFICIENCY

A common pathology in the nuclear fuel cycle industry is the sequence in which one organization designs a facility, another builds the plant with inevitable field changes, and yet a third cadre is charged with training operators and writing procedures as well as operating, maintaining, and changing the plant. This sequence all but ensures that important information will be lost. It is important that the organization charged with operating a facility have the technical talent to understand the process and its safety base. It is also important that an effective transfer of information occur on the assumptions for a safe operation.

I once worked at a facility with a solvent-extraction system that was designed by an architect–engineering firm, built by a construction contractor, and operated by a sequence of organizations. Part of this system was a three-stage decanter for a raffinate stream. As various operational problems occurred in the early days of the plant, the decanter plumbing was simplified to one stage. The various review groups approved the changes because the original design appeared to be overkill. As the staffing levels of the plant were expanded to support process analysis, it was determined that the original design was necessary for a subtle accident sequence. Until the defect was corrected, the facility was operating with an excessive risk of a criticality accident because of a communications breakdown. The problem was uncovered, however, because of the technical self-sufficiency of the operator.

A few isolated skill areas can be contracted out of the organization, but the basic technology of the nuclear fuel cycle and its safety base needs to remain within the capability of those responsible for the operation. The same is true for the various specialties within the safety disciplines. Plant operators must understand the assumptions that underlie criticality safety calculations. In a similar vein, those with operations responsibilities must understand the particular vulnerabilities of Failure Modes and Effects Analysis, Hazard and Operability Studies, or Probability Risk Assessments.

Another vital component of self-sufficiency is the ability to make changes. At one time I was asked to evaluate the corporate risk of assuming the management of a plutonium fuel facility. I noted that, without a plutonium critical mass laboratory, the ability to make innovative changes to the facility was seriously curtailed, which significantly affected the risk of managing the facility. Because the ability to make changes is a vital part of any safety program, the recent history of the United States critical mass laboratories is unsettling.

Eight years ago there were three general-purpose critical facilities. In 1988 the Nuclear Criticality Safety Division of the American Nuclear Society issued a white paper on the proposed closure of the Plutonium Critical Mass Laboratory at Hanford. It was noted in that paper that without the laboratory the industry would be handicapped in (1) responding adequately to new safety concerns with existing facilities and

(2) updating existing facilities that handle liquid plutonium with innovative designs that require new data. The laboratory was shut down, and the responsibility shifted to the two remaining facilities. More recently, one of the survivors went under and thus left a single facility with unstable support.

Good engineering practice and government regulations require that nuclear criticality safety in fissile material operations be based on experimental data. Coincident with the demise of experimental facilities, however, was the increasing ability to develop computer models of nuclear fuel cycle operating systems. The basis for safety has slowly switched to calculational results. This analytical capability has progressed far beyond the experimental database necessary to confirm the results. As a result, we need more benchmark experiments, not fewer. The vast majority of the existing benchmark critical experiments were not intended as benchmarks for computer studies and are therefore poorly documented for such a use. Good engineering practice and sound safety programs require critical mass laboratories for the nuclear fuel cycle.

In a larger sense, we need a nuclear safety culture that knows the physics and engineering of real systems that can and do go critical. A general-purpose critical facility is ideally suited for this chore. It is hard to imagine how such a culture can be nurtured without it: the training of nuclear criticality specialists is significantly enhanced by time spent at a critical mass laboratory getting hands-on experience with their craft.

FACING FACTS

In his report to GPU, Rickover affirmed that:

> Facing up to difficulties, regularly informing higher levels of management of problems and determining and correcting their root cause involve attitudes and practices which are essential to operating competence. Unfortunately, there is a disposition in all operating organizations to minimize the potential consequences of problems and to try to solve them with the limited resources available at the level where they are first recognized. The practice of forcing problems up to higher levels where greater resources can be applied must be assiduously fostered by top-level managers.[2]

The persistence of significant safety problems because of a reluctance to pass "bad news" up the authority chain is a common pathology in society. A review of criticality accidents in the nuclear fuel cycle exposes a common thread of hardware setups and operator practices that the management teams reported were unknown to them. Those with criticality safety responsibilities need to work toward a culture of effective communications; occasionally we need to force the issue. The Challenger accident is an example of technical information that was not transmitted upward.

We sometimes hear that delivering unwelcome information to responsible management of nuclear facilities can be a career-limiting action. Examples of "dead messengers" do exist. Some years back I was on an investigating committee with people from a wide range of prior responsibilities in the operations and safety of nuclear facilities. One of the issues of the investigation was the response of an organization to internal reporting of safety problems. As we discussed the issue, it became clear that most of us had experienced career setbacks for reporting safety problems. Further discussion revealed that none of the committee would have kept silent even had they anticipated the career risk. Given that the political consequences of honest reporting are a legitimate concern, however, the only professional response is to develop mechanisms for the "safe" upward transmittal of bad news. Falling on our sword can be only an occasional solution. One method I have used is to establish a broad-based committee with the charter of formally and annually reporting on the top ten criticality safety problems at the facility. The committee can address chronic safety issues. Also, broad committees are harder to intimidate than individuals. Other and possibly better methods should be explored.

The lack of support for critical mass experimental facilities is a community-wide consequence of the failure to communicate safety-related issues. If the issue of adequate method validation to support good safety evaluations had been made visible to project and operations officials, the critical mass laboratories might be thriving today. That they are not is an institutional failure of the community of criticality specialists.

RESPECT FOR RADIATION

On some sites criticality safety has the appearance of an academic exercise rather than true personnel protection. Fissile material handlers and safety specialists need periodic reminders that an accident can produce high, and potentially fatal, radiation levels. The accidents that have occurred in processing plants in the United States have had yields from 1.3×10^{17} to 4×10^{19} total fissions. The two criticality excursions that resulted in deaths had yields just over 10^{17} fissions. The unshielded lethal radius for a 10^{17} fission (3-MW/s) yield excursion is some 3 m. A 10^{19} fission (300-MW/s) unshielded excursion could have a lethal radius of 30 m, and a 10^{21} fission (30 000-MW/s) excursion could be deadly to 300 m. Higher yields (3×10^{22} fissions) have been hypothesized,[4–6] although many believe they are not credible. It is clear, however, that a criticality excursion at these yields would be a catastrophic event on any scale. The anticipated consequence in fatalities would increase from one to many dozens.

It has been said that a respected foe is a studied foe. Yet very little work has been done to support analytical models to predict accident yields. Such models could be used to disposition the super yield hypothesis. If such yields are possible, design features that could limit the yield could and should be studied with the models. Realistic models could be used to guide emergency planning. Some evidence suggests that the neglect of modeling is ending. It is encouraging to see the recent emergence of such codes as CRITEX from the United Kingdom, POWDER from France, and the calculational results from the University of Arizona. The NRC recently sponsored a student and some work with the SKINATH code from the University of Tennessee. The need for benchmark experiments to test the modeling is clear. Criticality safety specialists need new methods in order to effect real safety in real systems.

IMPORTANCE OF TRAINING

The Admiral considers that, "after the technical design of the plant itself, the most important element in assuring reliable and safe operation. . .is the training of the crew who will operate the plant."[2] He reviewed GPU training effort in the areas of facilities, staff, reporting level of trainers, and training requirements for entry level employees and operators. In particular, he noted that "top managers were directly involved with the training activities in observing classes, setting high standards, providing resources and monitoring the progress of the program to ensure its continued performance and improvement."

Criticality safety professionals have long appreciated the essential role of training, but we have not seen the commitment of resources that the Admiral thought necessary. We have seen the need for training of (1) all employees with unescorted access, (2) operators and maintenance personnel, (3) managers, and (4) criticality specialists themselves. Yet we seldom have the advantage of designing operator safety training, for example, at the same time that the process is being engineered. Criticality safety training is all too often an afterthought, even when the barriers to an accident are largely administrative. Deciding on the requirements for safety training during the design phase of a process would be more than a sobering and useful drill; it would significantly improve designs and help ensure adequate resources for training.

The practice at the Los Alamos National Laboratory of using the Critical Experiments Facility to train operators and fissile material handlers displays a commendable vision for the role of training. The fact that institutional support for this facility is volatile must be of concern to the whole community.

The training of criticality safety specialists is a longstanding issue. During my tenure on the Nuclear Criticality Technology and Safety panel, we designed an intern program; however, the program has seldom been used. This program entailed training in (1) computational methods with a methods development group in the DOE complex, (2) critical mass physics at a critical mass laboratory, and (3) criticality safety exposure at a complex fuel cycle facility. This training is actually minimal, but it is perceived as too costly by those with budget and schedule responsibilities. It would be interesting to hear Rickover's view of the state of criticality safety specialist training.

CONCEPT OF TOTAL RESPONSIBILITY

Safety comes from an effective integration of all the program elements in making operating decisions. Maintenance, technical support, quality control, radiological control, and criticality safety are essential operational elements and must be managed accordingly. Rickover noted that many of these elements were not performing well at the TMI plant before the 1979 accident, but that even if all the support functions were adequate, the integration was poor.

He noted fundamental changes in the 1983 organization and highlighted some management practices as evidence:

- The support service people made regular plant tours, including off shift, to see how their services and procedures were effective.
- Senior support function managers attended shift turnover meetings.
- Operations managers conducted briefings for shift crews.
- Accurate information was relayed to corporate headquarters.

All these examples are related to communication. Criticality safety specialists need to work on communication. Historically we have not done well. We assume we understand how process systems operate without talking to operators and other experts. We surmise our assumptions in evaluating a system or approving a process are understood without ensuring that it is so. We spend inadequate time walking around the facility, particularly on off shifts. We do not attend turnover meetings to learn of developing issues that affect us. Our professional meetings rarely address communications problems and solutions to them. All of us could learn to do a better job.

Some criticality safety groups use noncommunication as a deliberate strategy. Some years ago I reviewed a program in which it was the policy not to tell operations management the safety assumptions or margins of safety for the process operations because of the possibility that operations personnel would do something sneaky with the information. I recall my stunned reaction as a facility manager told me he could not tell me the margin of safety for a particular operation because he was not permitted to know, but he volunteered to leave the area so that a safety specialist could tell me. Innovations in safety programs are welcome, and it would be tragic if all programs looked the same. Good safety analysis and effective procedures, however, depend on informed input from all involved parties. Programs with built-in mistrust and noncommunication are extremely fragile.

CAPACITY TO LEARN FROM EXPERIENCE

The Admiral asserted that "a capacity to acknowledge mistakes and to search out and correct their underlying causes is essential to nuclear operations." The essence of maturity for an individual or an organization is the ability to learn the right lesson from experiences. If we do not wish to be menaced by immature organizations running nuclear facilities, we need to ensure that we and our organizations can learn from experience.

As an industry, we have profited from studying the historic criticality accidents. Many of the American National Standards Institute standards on proper practices in criticality safety are based on lessons learned from these. We need, however, to continue to reflect on these experiences because the root causes of these accidents still plague our facilities. I recommend each year identifying the root causes of at least one of the historic accidents and evaluating our own facility against the lessons learned.

We have gained a great deal from pondering the published accidents. It is intriguing to speculate what we could gain from the unpublished ones. The recent opening of the former Soviet Union to communication raises the possibility of greatly expanding the available accident database. A Russian official visiting the NRC referred to 12 such accidents. Some of these accidents are beginning to circulate as oral history, such as the accident in a shielded hot cell hood in which the radiation effects of a criticality excursion cost the worker both arms. Government agencies and we as a community should continue to pursue this treasure load of potential data. We need to study the accidents with mature reflection and assess our own facilities for lessons learned.

The lessons learned from the accident that occurred on my watch at the Idaho Chemical Processing Plant (ICPP) have been etched into my consciousness. We learned a great deal about the shortcomings of our organization:

- Weaknesses in our configuration control system: control devices on plant drawings were not in place.
- Process controls that were used for safety purposes without the added rigor necessary.
- Weaknesses in our document control system: out-of-date procedures and run sheets had been copied and were in use.
- Subtle process sensitivities important to safety: computer code models were acquired or developed to address this problem in our solvent-extraction system.
- Inappropriate levels of operator training: the average experience level of operators had decreased, but we did not respond with better training.

These and other lessons from our experience have been shared widely in the criticality safety and fuel-cycle communities, yet these weaknesses appear surprisingly often in other production facilities I have visited in the intervening years. We need to reinforce the concept of learning from the experience of others. It is much less painful and vastly cheaper, although apparently not as effective.

A more plentiful source of experience is events at our own facility. At the ICPP we used a formal incident investigation methodology to get a consistent benefit from reviewing incidents. It started with prompt notification, continued with prompt meetings with the involved staff to get as much accurate data as possible (including holding over operators if the event happened late on the shift), and included, if appropriate, a root-cause investigation. Events are a rich source of basic information on plant problems, and they can generate corrective action that significantly improves both the operating efficiency and plant safety. Most significant plant events are caused by a multitude of weaknesses in processes and administrative systems. A collection of these events will uncover a cross section of the pathology of a plant.

The ICPP once had an unused room or cell that was being characterized for decommissioning. We got quite excited when we found many kilograms of uranium in one of the large tanks in the cell because the uranium concentration seemed to be rising toward an unsafe level; however, that proved to be a result of deviations among successive samples. This shielded cell had been operated remotely from an instrument panel in the operating area. The instruments were not maintained for this obsolete cell, and they became unattractive. The unattractive instruments were then removed during a cleanup effort and not replaced. Without instruments, we were blind to activity in the cell, even though none was considered possible. During a facility modification after the instrument removal, a process pipe with communication to the cell was briefly connected. At the same time we had an apparent inventory loss of uranium. We used the statistical uncertainties of measurements to resolve the accountability problem and replumbed the facility to the normal status. The investigation committee determined that the uranium had entered the cell at the time of modification years before when the cell was peripherally used for a few days. Uranium in unapproved locations is normally a significant criticality safety problem. The lessons learned included the following: (1) unused equipment continues to need instruments, (2) expanded operational monitoring is required during maintenance or construction, (3) sending and receipt volume logs for fissile solutions are necessary when the possibility of loss exists, and (4) the safety problem and the accountability problem are different and require different solutions. We had formal mechanisms to evaluate and appropriate the tutoring of experience into changed practice. These lessons and others affected subsequent plant operations.

The proverb "Those who don't remember the lessons of history are doomed to repeat them" applies to nuclear facility safety and summarizes Rickover's final criterion for competence.

CONCLUSIONS

In summary, it may be useful to restate Rickover's seven criteria in the language of criticality safety:

- A criticality safety group should have a broader vision than regulatory requirements and should ensure a rising standard for adequacy.
- A site with fissile material needs a staff that well understands the technical basis for the operations and its safety assumptions in order to sustain a competent program.
- Recognition of individual and institutional aversion to "facing facts" should lead to developing coping mechanisms to force resolution of criticality safety issues.
- Analytic modeling of criticality excursions should be supported to aid in safer system designs and better emergency response programs.
- Criticality safety training should be a driving function of fissile material operation from equipment design to decommissioning.
- No safety program will work well without good communication between operations, maintenance, and the various safety disciplines. All the programs must work and work together.
- Mature safety programs have a willingness and a disciplined approach to learn from experience.

Most of us are caught up in the press of problems, events, and deadlines. We need to ponder periodically where we and our organizations are going. We need to ask how competent are the elements of our programs and how do they effect criticality safety. Rickover's checklist is a valuable reality check as we strive for competence or even excellence.

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Accident Analysis

Edited by R. P. Taleyarkhan

Transient Analysis of the PIUS Advanced Reactor Design with the TRAC-PF1/MOD2 Code[a]

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Abstract: *The PIUS Advanced Reactor is a 640-MW(e) pressurized-water reactor developed by Asea Brown Boveri. A unique feature of the PIUS concept is the absence of mechanical control and shutdown rods. Reactivity normally is controlled by the boron concentration in the coolant and the temperature of the moderator coolant. Analyses of five initiating events have been completed on the basis of calculations performed with the system neutronic and thermal–hydraulic analysis code TRAC-PF1/MOD2. The initiating events analyzed are (1) reactor scram, (2) loss of off-site power, (3) main steam-line break, (4) small-break loss of coolant, and (5) large-break loss of coolant. In addition to the baseline calculation for each sequence, sensitivity studies were performed to explore the response of the PIUS reactor to severe off-normal conditions having a very low probability of occurrence. The sensitivity studies provide insights into the robustness of the design.*

The PIUS Advanced Reactor is a four-loop, Asea Brown Boveri (ABB)-designed pressurized-water reactor (PWR) with a nominal core rating of 2000 MW(t) and 640 MW(e).[1] The fuel rods and assemblies are similar to those in modern PWRs; however, the assembly height is approximately 60% of modern PWR plants. A schematic of the basic PIUS reactor arrangement is shown in Fig. 1. The schematic generally is representative of the design except that the downcomer and riser are integrated rather than separated, as shown in the schematic. Reactivity is controlled by the boron concentration in the coolant and by temperature; there are no mechanical control or shutdown rods. The core is submerged in a large pool of highly borated water and is in continuous communication with the pool water through pipe openings called density locks. These locks provide a continuously open flow path between the primary system and the reactor pool. The reactor coolant pumps (RCPs) are operated so that there is a hydraulic balance in the density locks between the primary system and the pool; thus the pool water and primary coolant are kept separate during normal operation. Hot primary system water is stratified stably over cold pool water in the density locks. PIUS contains an active-scram system, which consists of four valved lines (one for each primary coolant loop) that connect the reactor pool to the inlets of the RCPs. Although the active-scram piping and valves are safety-class equipment, operation of the nonsafety-class RCPs is required for effective delivery of pool water to the primary system. PIUS also has a passive scram system; this will function if one or more of the RCPs loses its motive power and thus eliminates the balance between the primary system and the pool and activates flow through the lower and upper density locks. In addition, the balance cannot be maintained after the RCP overspeed limit of 115% is reached. In all such cases (i.e., loss of RCP motive power or reaching the RCP overspeed limit), highly borated water from the pool will enter the primary system via natural circulation, which will shut down the reactor and cool the core.

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[b]Technology and Safety Assessment Division, Los Alamos National Laboratory.

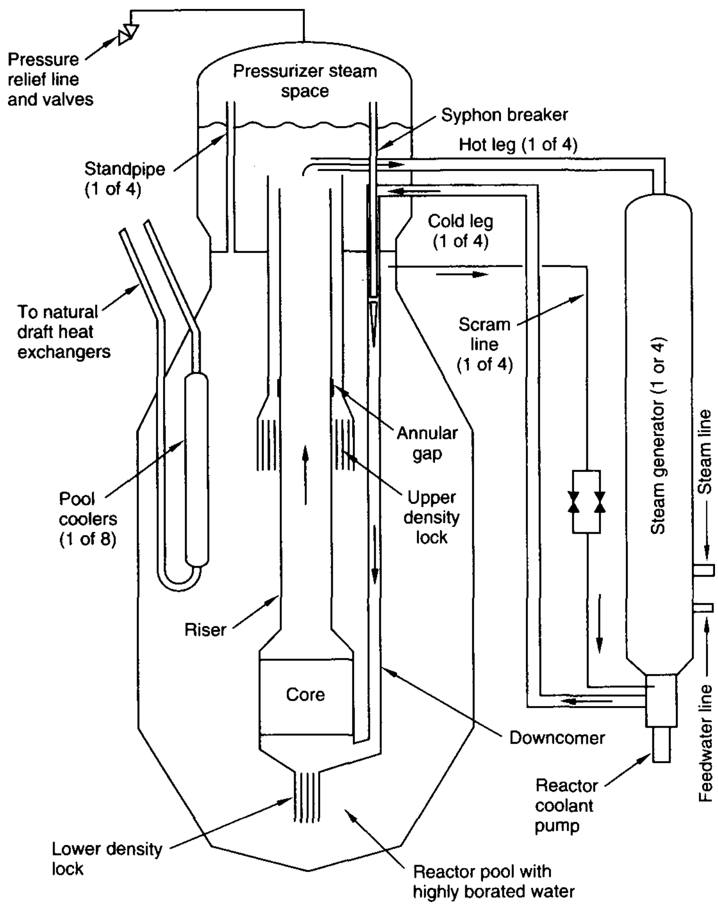

Fig. 1 Schematic of the PIUS reactor.

The heated coolant will return to the pool, which can be cooled by either an active, nonsafety-class or fully passive, safety-class system. In reference to Fig. 1, the path of the natural-circulation flow is as follows: pool—lower density lock—core—riser—annular gap—upper density lock—pool.

As part of the preapplication and eventual design certification process, applicants for certification are required to submit neutronic and thermal–hydraulic safety analyses over a sufficient range of normal operation, transient conditions, and specified accident sequences. ABB submitted a Preliminary Safety Information Document (PSID)[2] to the United States Nuclear Regulatory Commission (NRC) for preapplication safety review in 1990. Early in 1992, ABB submitted a Supplemental Information Package to the NRC to reflect recent design modifications.[3] An important feature of the PIUS supplemental design was the addition of the previously described active-scram system as the first-line shutdown system for most transient and accident conditions. This system cannot meet all scram requirements because its performance depends on the operation of the RCPs; therefore the passive scram system of the original PSID design was retained. Because PIUS does not have the usual rod-based shutdown systems, the response of PIUS following both planned reactor trips and a variety of accident initiators must be examined carefully and understood.

The PIUS safety analyses submitted by ABB are based on results from the RIGEL code,[4] a one-dimensional (1-D) thermal–hydraulic system analysis code developed at ABB Atom. The review and confirmation of the ABB safety analyses for the PIUS design constitute an important activity in the NRC's preapplication review. Safety analyses use applicable criteria (e.g., 10 CFR 50.46 and associated Appendix K) for evaluating the performance of emergency core-cooling systems. Los Alamos supported the NRC's preapplication review of the PIUS reactor. This article summarizes the results of analyses performed to understand the response of the PIUS supplemental design to each of the following five baseline events:

1. Reactor scram
2. Loss of off-site power (LOSP)
3. Main steam-line break (MSLB)
4. Small-break loss-of-coolant accident (SBLOCA)
5. Large-break loss-of-coolant accident (LBLOCA)

In addition to analyzing each baseline sequence, sensitivity studies were performed to explore the robustness of the PIUS concept to severe off-normal conditions with a very low probability of occurrence. All calculations performed with the Transient Reactor Analysis Code (TRAC)-PF1/MOD2 were best estimates (i.e., nominal design power limits and setpoint limits were modeled).

TRAC ADEQUACY FOR THE PIUS APPLICATION

Version 5.3.05 of TRAC was used for each calculation. The TRAC series[5] was developed at Los Alamos National Laboratory (LANL) to provide advanced, best-estimate predictions for postulated accidents in PWRs. The code incorporates four-component (liquid water, water vapor, liquid solute, and noncondensable gas), two-fluid (liquid and gas), nonequilibrium modeling of thermal–hydraulic behavior. TRAC features flow-regime-dependent constitutive equations, component modularity, multidimensional fluid dynamics, generalized heat structure modeling, and a complete control systems modeling capability. The code also features a three-dimensional (3-D), stability-enhancing, two-step method that removes the Courant timestep limit within the vessel solution. Finally, a higher order (second) Godunov method for solute tracking is available that reduces numerical diffusion significantly. Many of these features have proved useful in modeling the PIUS reactor.

Code adequacy must be addressed when first applying a computer code to a new reactor type (e.g., PIUS). If TRAC analyses supported a design certification activity, a formal and structured code-adequacy demonstration would be needed. One such approach would be to (1) identify representative PIUS transient and accident sequences; (2) identify the key systems, components, processes, and phenomena associated with the sequences; (3) conduct a bottom-up review of the individual TRAC models and correlations; (4) conduct a top-down review of the total or integrated code performance relative to the needs assessed in steps 1 and 2; and (5) correct significant identified deficiencies. The bottom-up review determines the technical adequacy of each model by evaluating its pedigree, applicability, and fidelity with the use of fundamental, separate-effects, or component data. The top-down review determines the technical adequacy of the integrated code by evaluating code applicability and fidelity with the use of integral test facility data.

Because the NRC is engaged in a preapplication rather than a certification review, the NRC and LANL concluded that a less extensive demonstration of code adequacy would suffice. Steps 1 and 2 were performed and documented.[6] A bottom-up review specific to the PIUS reactor was not conducted. The bottom-up review of TRAC conducted for another reactor type,[7] however, provided some confidence that many of the basic TRAC models and correlations are adequate, although some necessary code modifications also were identified. A complete top-down review was not conducted. The ability of TRAC to model key PIUS systems, components, processes, and phenomena was demonstrated in an assessment activity[8] with the use of integral data from a large test loop facility (ATLE).[4] ATLE is a 1/308-volume-scale integral test facility that simulates the PIUS reactor. Key safety features and components are simulated in ATLE, including the upper and lower density locks, reactor pool, pressurizer, core, riser, downcomer, reactor coolant pumps, and steam generators. Key processes are simulated in ATLE, including natural circulation through the upper and lower density locks, boron transport into the core (simulated with sodium sulfate), and control of the density lock interface. Core kinetics are simulated indirectly through a point-kinetics computer model that calculates and controls the core power on the basis of the core solute concentration, coolant temperature, and heater rod temperature. The results of this assessment activity will be discussed at the appropriate point in this article. The ability of TRAC to model key PIUS systems, components, processes, and phenomena was demonstrated further by benchmarking TRAC to the RIGEL code. The results of three benchmark comparisons also will be discussed at appropriate points.

TRAC includes the capability for multidimensional modeling of the PIUS reactor. A multidimensional thermal–hydraulic model has been prepared and used to calculate the baseline pump-trip scram and MSLB transients[9] for the original PSID design and LBLOCA[10] for the PSID supplemental design. The multidimensional LBLOCA results are summarized briefly in this article. The 1-D model is believed to represent many PIUS transients and accidents adequately with the following important reservation. The most important physical processes in PIUS are related to reactor shutdown because the PIUS reactor does not contain control and shutdown rods. Combined core neutronic and thermal–hydraulic effects may occur in PIUS, including multidimensional interactions arising from nonuniform introduction of boron across the core. ATLE does not simulate multidimensional effects. The RIGEL thermal–hydraulic model is 1-D, and a point-kinetics model is used. Although both 1-D and multidimensional TRAC thermal–hydraulic models have been used for PIUS analyses, core neutronics are simulated with a point-kinetics model in each case. The point-kinetics model implies that the entire core becomes subcritical at the same time, whereas a spatial-kinetics model would show a core power decrease beginning at the point where the boron is injected. If no positive reactivity is inserted concurrently, the absolute power density should not increase anywhere in the core, although the relative power distribution will show sharp axial gradients as the boron passes through the core. It is not known whether the results from the point-kinetics model are consistently nonconservative or conservative or if the conservatism or lack thereof varies for each transient analysis. Combined multidimensional core neutronic and thermal–hydraulic effects are believed to be important and should be investigated thoroughly if the design and safety review effort continues.

TRAC MODEL OF THE PIUS REACTOR

Descriptions of the TRAC multidimensional model of the original PSID design and the fully 1-D model of the PSID supplemental design are provided in Refs. 9 and 10, respectively. Because this article presents results primarily from the 1-D model, a brief description of this model is provided here. The four-loop TRAC model consists of 74 hydrodynamic components (727 computational cells). The reactor vessel comprises 16 components, each coolant loop comprises 8 components, and the remaining 26 components represent the pool, steam dome, density locks, and pressurizer line. One heat-structure component is used to represent the average fuel rods. The hot rod is modeled as an auxiliary rod (i.e., the hot rod carries the maximum power but is exposed to the core-averaged, thermal–hydraulic channel). The noding diagram for the TRAC-PF1/MOD2 1-D model of the PIUS vessel and pool is shown in Fig. 2, and the 1-D model of one of the PIUS coolant loops is shown in Fig. 3. The TRAC 1-D model is noded more finely than the RIGEL model because of Los Alamos' modeling preferences; however, no particular merit is attributed to the finer noding. The TRAC-calculated and PSID supplemental steady-state values are tabulated in Table 1 for comparison.

Additional initial and boundary conditions for the calculated transients generally are as follows except where otherwise noted. The reactor is operating at the beginning of cycle (BOC) with 100% power and a primary loop boron concentration of 375 ppm. The boron concentration in the reactor pool initially is 2200 ppm. If the active-scram system is activated, the scram valves open over a period of 2 s following event initiation, remain open for 180 s, and close over a period of 20 s. The feedwater pumps supplying coolant to the steam generator secondary side are tripped at the time of reactor trip, and the feedwater flow rate decreases linearly to zero in 20 s. The steam drum pressure on the steam generator secondary side is kept constant at 3.88 MPa. The RCPs have an overspeed limit of 115%.

In the following sections the results for five types of initiating events are presented. Results are summarized for each baseline transient and the associated sensitivity studies. The sensitivity studies focus on two important processes[6] (core shutdown and core cooling) by postulating low-probability sequences that compromise normal shutdown and cooling. In each case the fully 1-D model was used. In addition, the 3-D model was used for the baseline LBLOCA analysis. Only brief descriptions of the comprehensive results are possible in this summary paper. Additional details are provided in Refs. 10 to 14. When applicable, the results of TRAC assessment and benchmarking activities are presented with the use of data from the ATLE facility and comparisons of TRAC- and RIGEL-calculated results for the same transient.

REACTOR SCRAM EVENTS

The active-scram system was incorporated in the PSID supplemental design with the intent that it would function for most anticipated and accident transients. The baseline active-scram transient is initiated by opening a valve in each scram line connecting the reactor pool to the RCP inlet. Essentially all important

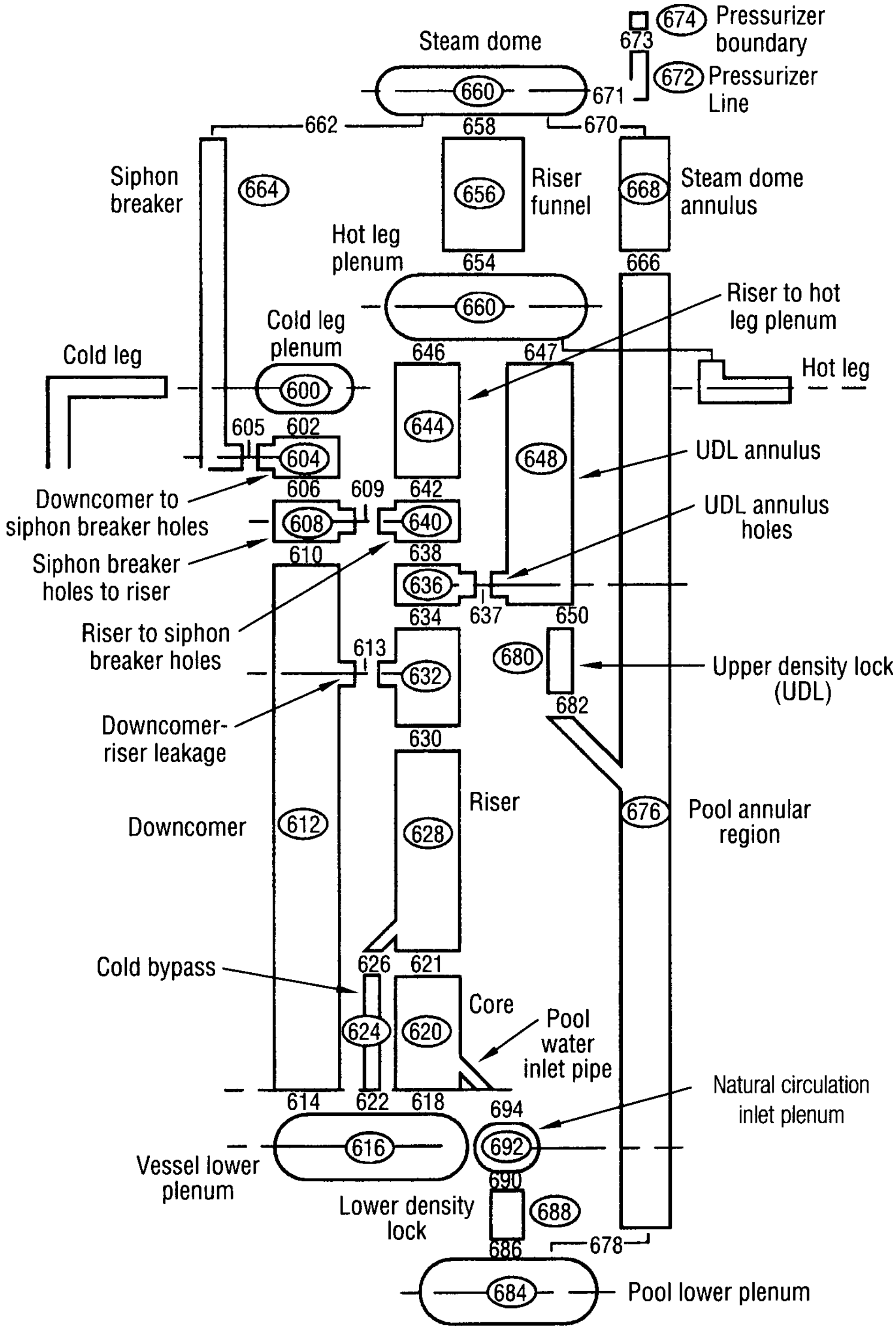

Fig. 2 TRAC 1-D model of the PIUS reactor vessel and pool. Circled numbers represent components in model. Other numbers represent junction numbers.

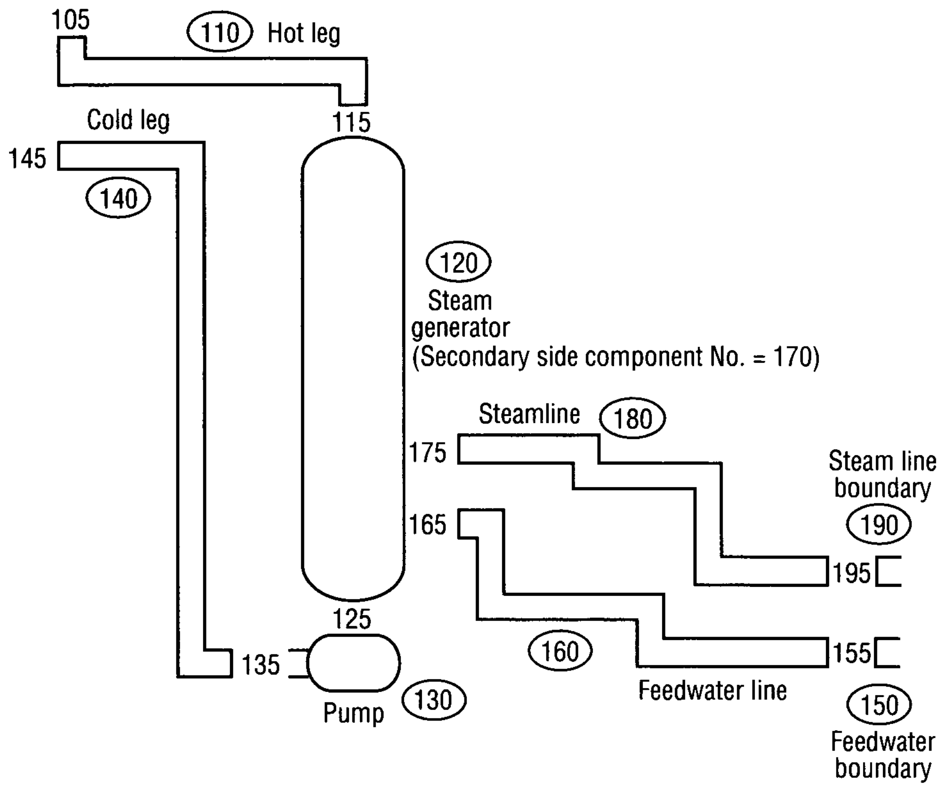

Fig. 3 TRAC 1-D model of a PIUS coolant loop. Circled numbers represent components in model. Other numbers represent junction numbers.

Table 1 TRAC-Calculated and PSID Supplemental Steady-State Values

Parameter	TRAC	PSID supplement
Core mass flow, kg/s	12 822	12 880
Core bypass flow, kg/s	200.2	200
Loop flow, kg/s	3 266	–
Cold-leg temperature, K	531	527.1
Hot-leg temperature, K	560.7	557.3
Pressurizer pressure, MPa	9.5	9.5
Steam exit pressure, MPa	4.0	4.0
Steam exit temperature, K	540.3	543
Steam flow superheat, K	15.3	20
Steam and feedwater mass flow, kg/s	243	243

phenomena arise from opening the scram valves and terminating feedwater flow to the steam generators. The total scram line flow, which varies between 700 and 800 kg/s, produces several effects. First, primary coolant is displaced from the primary system and enters the reactor pool, primarily through the upper density lock but also through the lower density lock (Fig. 4). Second, the highly borated water injected by the active-scram system mixes with the primary coolant. The boron concentration increases rapidly when the scram valves are open; however, the increase is terminated when the scram valves are shut. The core-inlet primary boron concentration stabilizes at approximately 860 ppm (Fig. 5). The increasing concentration of boron in the core inserts sufficient negative reactivity to reduce the core power to decay-heat levels (Fig. 6). Following closure of the scram valves, the flows of highly borated pool water through the active-scram system into the primary system are terminated, and control of the lower density-lock thermal interface is recovered by the RCP speed control system. Primary-to-secondary heat transfer in the steam generators terminates by 115 s following the early trip of the main feedwater pumps. Thus core decay heat is deposited in the primary coolant, and fuel and coolant temperatures begin a steady 40-K/h increase. If no action is taken, the primary will continue to heat, the RCPs will increase in speed to maintain control of the lower density-lock

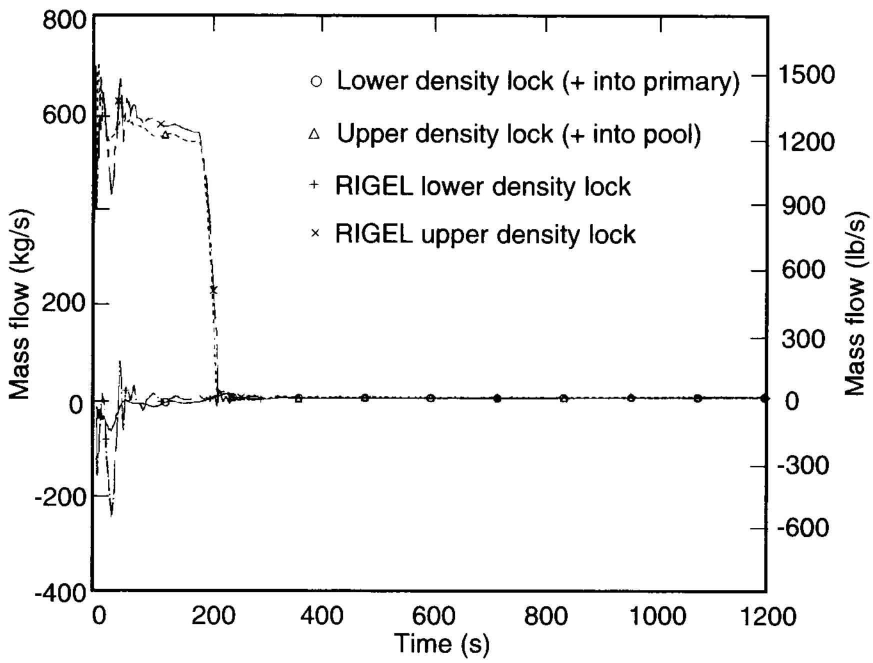

Fig. 4 Density-lock flows for active-scram-system baseline case.

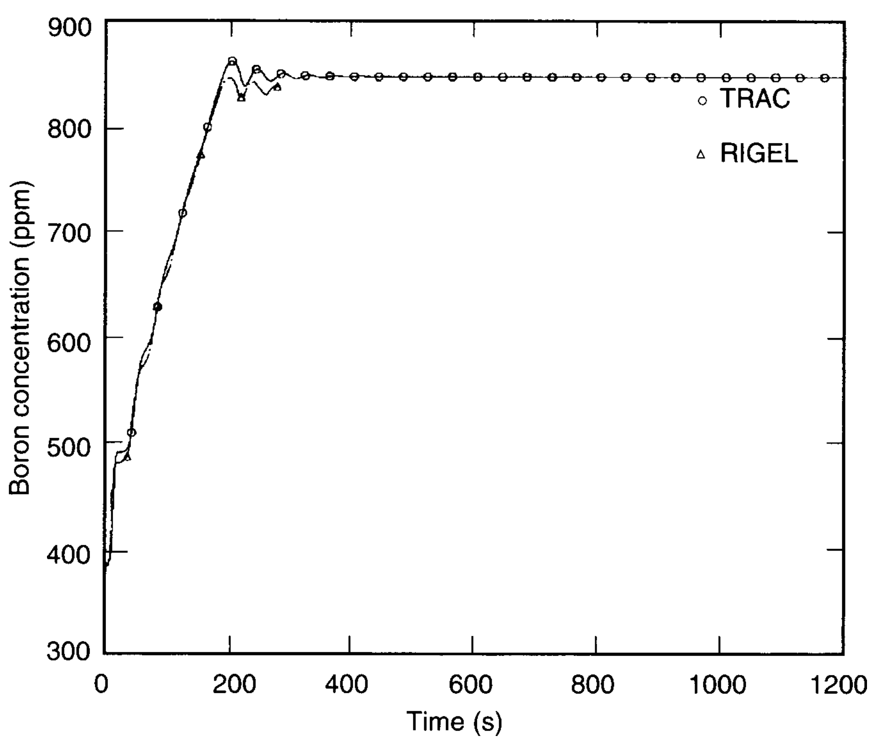

Fig. 5 Primary boron concentration for active-scram-system baseline case.

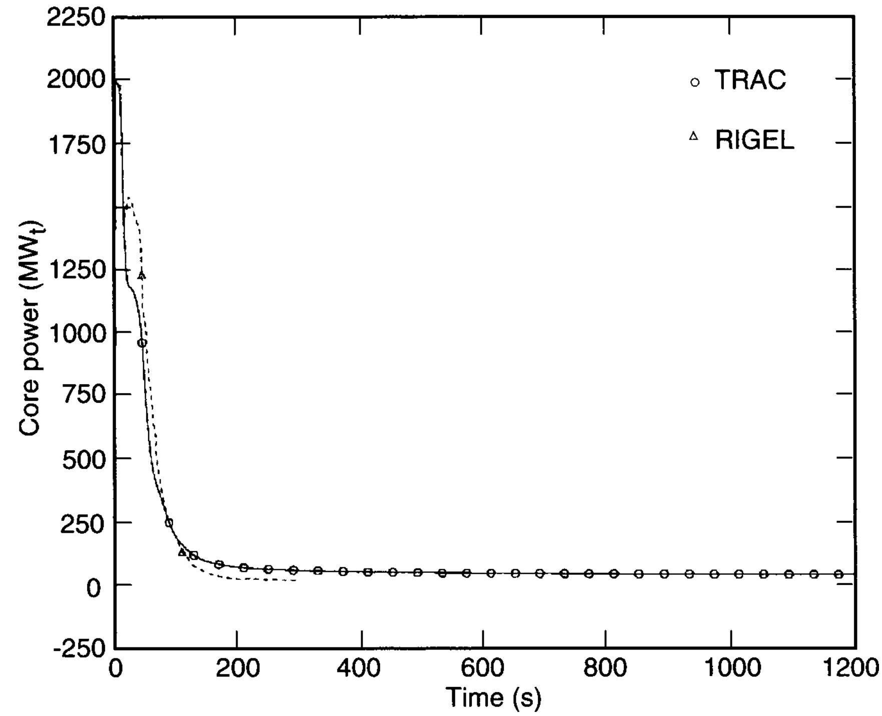

Fig. 6 Core power for active-scram-system baseline case.

thermal interface until the RCP overspeed limit of 115% is reached, and the density locks will activate to initiate natural circulation between the primary system and the reactor pool. The pool will be cooled either by active (nonsafety-grade) or passive (fully safety-grade) pool-cooling systems that reject core decay heat to the ultimate heat sink.

A RIGEL calculation of the active-system scram transient has been reported.[3] Several results from the RIGEL calculations have been coplotted with the TRAC-calculated results for this transient. The RIGEL calculations were terminated at 300 s, whereas the TRAC calculations were terminated at 1200 s. The TRAC- and RIGEL-calculated core powers are shown in Fig. 6. The upper and lower density-lock flows are compared in Fig. 4, and the primary-loop boron concentrations at the core inlet are compared in Fig. 5. The TRAC- and RIGEL-calculated results are both qualitatively and quantitatively similar and therefore are in reasonable agreement. Because the two codes were developed independently, this reasonable agreement provides added assurance that the major trends and processes associated with the active scram are represented correctly to the extent that they are well modeled by 1-D thermal hydraulics and point kinetics.

Sensitivity studies were performed to explore the robustness of the PIUS concept to severe off-normal conditions following active system trips. The most severe of these conditions are very-low-probability events. Fractional and complete blockages of the lower density lock were analyzed. Given the small flows through the lower density lock for the baseline transient, even a total blockage would produce only a minimal impact on the course of the transient. As a further assessment of the robustness of the PIUS concept, total blockages of both the upper and lower density locks were assumed. A shutdown in core power again is achieved. With both density locks blocked, the amount of pool water injected through the scram lines is reduced when compared with the baseline because primary inventory can be displaced into the reactor pool only through the small standpipes that connect the pressurizer steam space and the reactor pool (Fig. 1). With the reduced scram-line flow, the primary boron concentration increases to only 480 ppm before the scram valves close. For this transient, the core power decreases more slowly than in the baseline, and the fuel and moderator temperatures remain higher. Later in the transient, the increasing moderator temperature results in the largest negative reactivity contribution to the total reactivity. Other sensitivity calculations were performed to examine the effect of a reduced boron concentration. Active scrams with boron concentrations of 1800 and 1000 ppm were examined. The first concentration corresponds to the level at which a reactor scram is initiated on a low boron concentration.[3] The second concentration corresponds to the condition at which a critical core can be achieved at cold shutdown conditions and BOC. For the 1800-ppm case, core power decreases at a slightly slower rate than the baseline; however, the power levels are indistinguishable by 200 s. The active system scram with the boron concentration at 1000 ppm also culminates in a shutdown condition, although the phenomena are markedly different. The core power decreases at a slower rate than in the baseline and does not reach the same level as the baseline until 400 s. Consequently the extra decay heat deposited in the primary system causes the system to heat and pressurize. The pressure-relief-system safety valves open several times while the scram valves are open and open periodically after the scram valves are closed. Flow from the pool enters the primary system through the lower density lock and returns to the pool through the upper density lock. The pool is cooled by the available pool-cooling systems. Additional actions are required to terminate this event fully (e.g., injection of additional boron into the primary system).

LOSP EVENTS

An LOSP transient demonstrates the passive-scram function of the PIUS reactor. With the loss of motive power to all RCPs, the pumps coast down and the active-scram system becomes unavailable. The passive scram is associated inherently with the LOSP. The steam generators dry out by 70 s, after which primary-to-secondary heat transfer is terminated. The hydraulic balance in the density locks between the primary coolant loop and the pool is upset with the loss of the RCPs. There is a rapid inflow of water into the primary system through the lower density lock and a corresponding but lower flow from the primary system back to the reactor pool through the upper density lock (Fig. 7). The difference between the two flows is caused by the volumetric shrinkage of the primary system coolant as fluid temperatures decrease. The lower density-lock flow peaks at 1225 kg/s, shortly after LOSP initiation, and decreases thereafter until the flow rate required to remove core

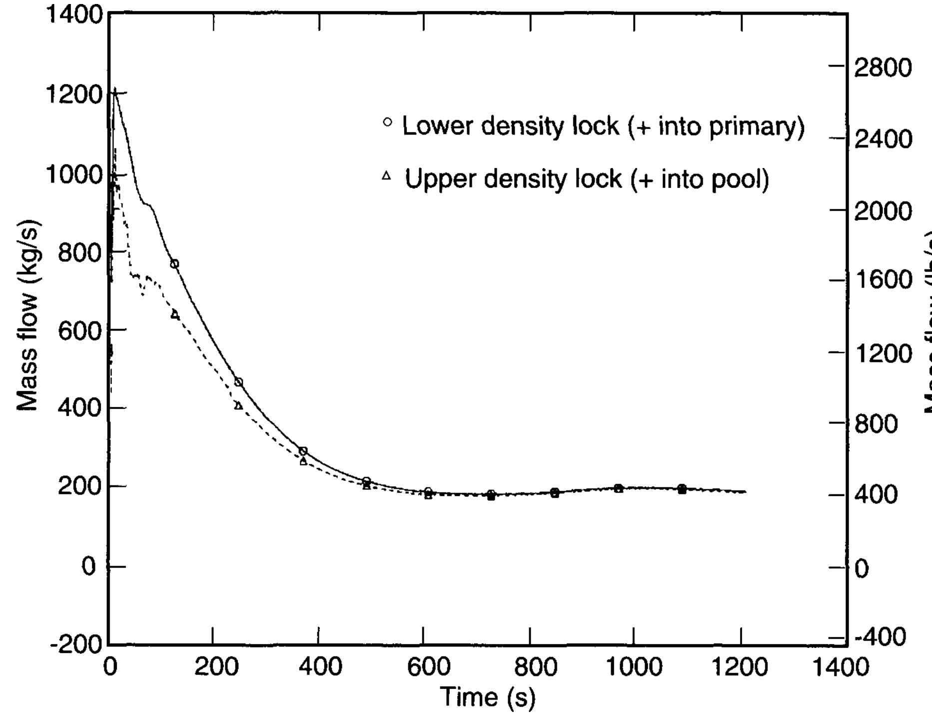

Fig. 7 Density-lock flows for the LOSP baseline case. LOSP is loss of off-site power.

decay heat (~200 kg/s) is established. The large influx of water passing from the reactor pool into the primary system through the lower density lock rapidly increases the primary system boron concentration to the pool concentration of 2200 ppm and lowers coolant temperatures at the core inlet to the pool coolant temperature of 323 K. The rapid decrease in fuel and coolant temperatures inserts positive reactivity. The negative reactivity inserted by boron is larger than the positive contributions; thus the total reactivity is negative (Fig. 8). The core power decrease to the decay-heat levels following an LOSP is more rapid than that following an active scram because the flow of borated pool water through the lower density lock is larger than the total flow through the active-scram system.

Sensitivity studies were performed to explore the robustness of the PIUS design to very-low-probability combination events following an LOSP. Calculations were performed to examine the effect of lower density-lock blockage fractions of 75 and 100%. For the 75% blockage case, the peak lower density-lock flow of 450 kg/s compares to a peak flow of 1225 kg/s for the baseline (unblocked) transient. This has several consequences: (1) the rate at which boron is introduced into the core is slowed; (2) the core inlet boron concentration increases to the pool value of 2200 ppm approximately 100 s later than in the baseline; (3) the core inlet temperature decreases to the pool temperature; and (4) the core outlet average coolant temperature reaches the saturation temperature shortly after the start of the transient, and there is a brief period of voiding in the core. The core-average voiding approaches 5%; however, it lasts only a few seconds, and there is no core dryout. The core power decrease to decay-heat levels is only slightly slower in the blockage case. The same decay-heat core power levels are reached after approximately 100 s.

Although the complete blockage of the lower density lock is a very challenging transient in regard to phenomena, PIUS successfully accommodates this very-low-probability combination transient. The density-lock flows are shown in Fig. 9. The lower density lock is completely blocked. The upper density lock is open to the reactor pool, and the interface is agitated for the first 375 s. The net flow from the primary system to the pool is negligible (~1000 kg), however. Because the active-scram system does not function when the RCPs are inoperable and there is little flow from the reactor pool to the primary system through the upper density lock, the dominant negative reactivity is

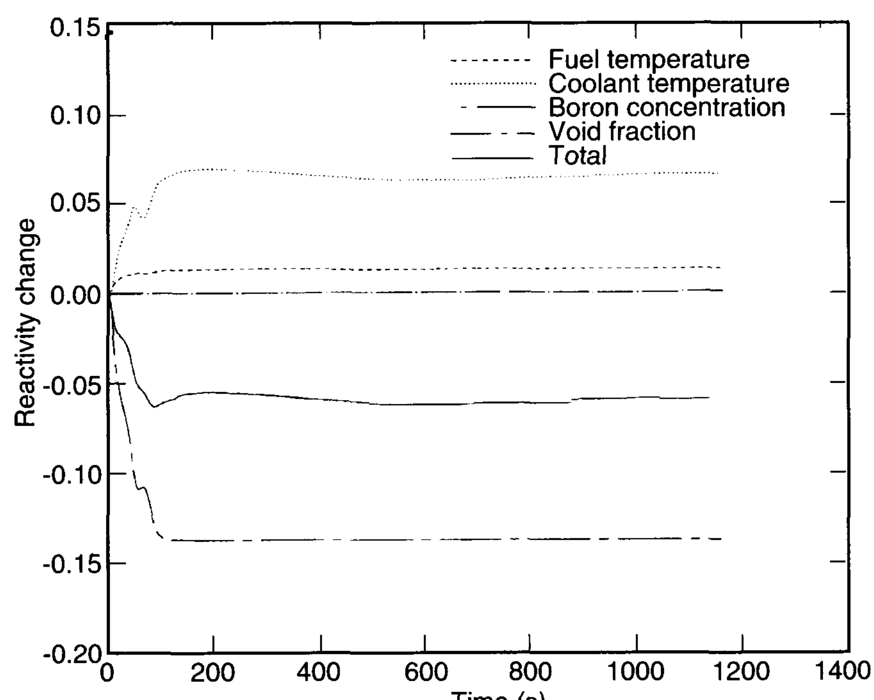

Fig. 8 Core reactivity changes for the LOSP baseline case. LOSP is loss of off-site power.

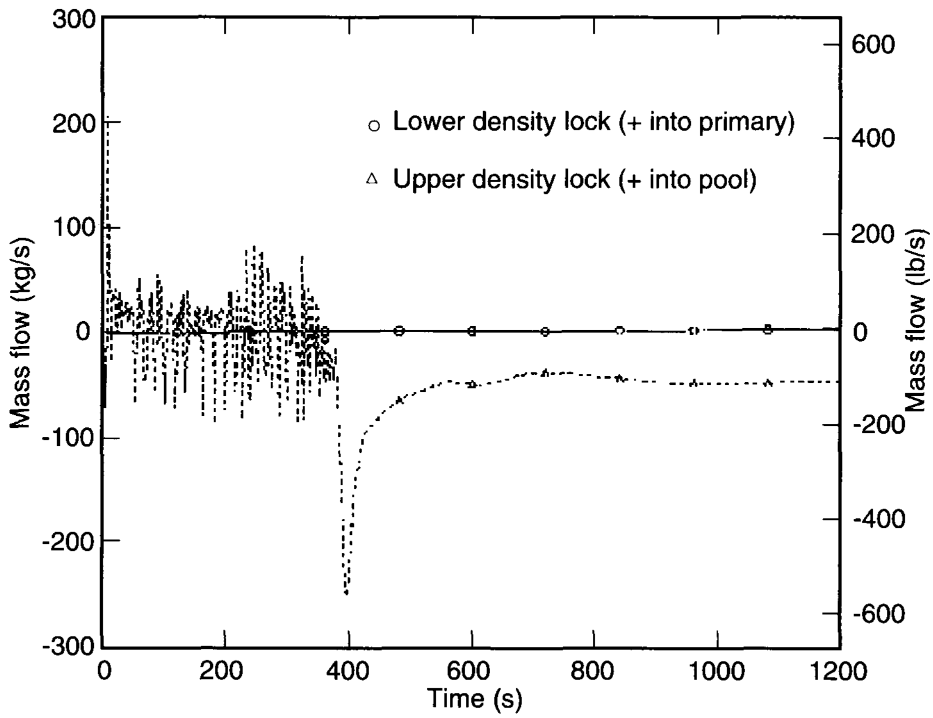

Fig. 9 Density-lock flows for an LOSP with complete blockage of the lower density lock. LOSP is loss of off-site power.

inserted early in the transient via the coolant temperature (moderator) and voiding (Fig. 10). The core power is reduced but remains more than 500 MW(t) until 200 s (Fig. 11). During this interval the primary system pressurizes and heats up. The safety valves open repeatedly after the opening setpoint of 12.3 MPa is reached. Primary-to-secondary heat transfer continues until the steam generators dry out at 235 s. The core inlet temperature increases rapidly following steam generator secondary dryout, and the increasing moderator temperature inserts sufficient negative reactivity to reduce the power further. Some voiding occurs in the core and peaks at slightly less than 7%. There is neither a core dryout nor a cladding temperature excursion. At 375 s, the upper density lock activates, and a natural-circulation flow from the primary system to the reactor pool starts in the upper portion of the reactor by way of the pressurizer standpipes (Fig. 12). By 600 s, a stable primary system flow circulation has been established. This circulation consists of a primary and secondary circulation. The primary circulation follows the normal flow through the primary loops. The secondary circulation is the means by which boron from the reactor pool enters the primary system. With the lower density lock blocked, this natural-circulation flow path varies from the normal natural-circulation path; the altered flow path is pool—upper density lock—annular gap—riser—pressurizer—standpipes—pool. Flow directions through the upper density lock and annular gap are reversed relative to the normal natural-circulation flow direction. The flow through the upper density lock matches the primary coolant that flows through the standpipes. The flow entering the upper density lock merges with a larger recirculation flow passing downward through the upper density-lock annulus. The combined flow passes into the riser through the overlapping joint (annular gap) between the riser and the upper density-lock annulus (Fig. 12).

Additional sensitivity calculations were performed to examine the effect of boron concentrations of 1800 and 1000 ppm. The differences between the calculated baseline and the 1800-ppm pool concentration case are small. The core power decreases at a rate only slightly slower than in the baseline and thus successfully accommodates an LOSP with a boron concentration in the pool water of 1800 ppm. The phenomena of the LOSP transient with a boron concentration of 1000 ppm are markedly different. The lower and upper density-lock flows are similar to those in the baseline; however, the core inlet boron concentration can

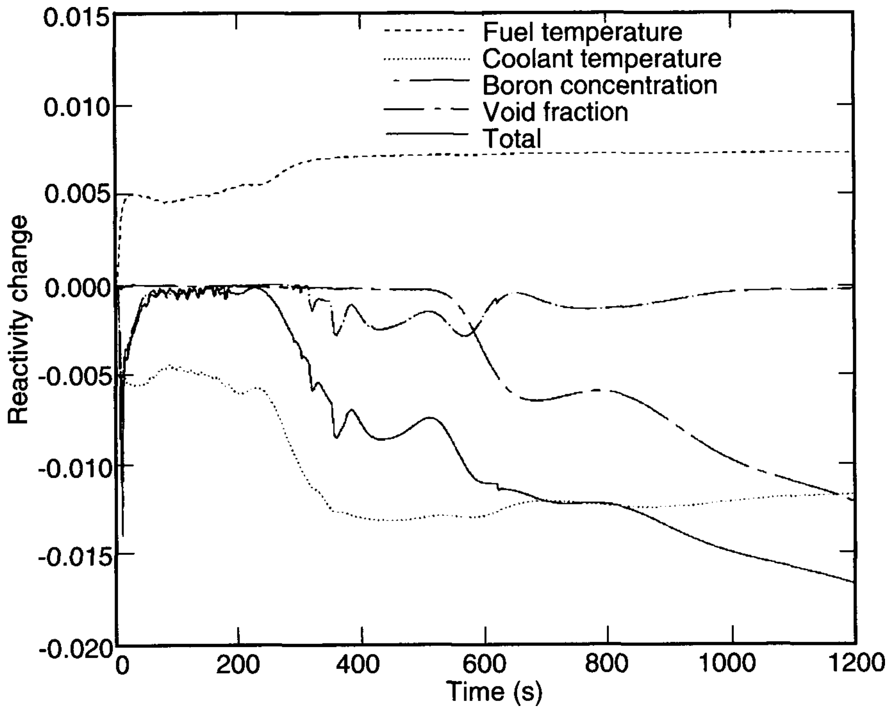

Fig. 10 Core reactivity changes for an LOSP with complete blockage of the lower density lock. LOSP is loss of off-site power.

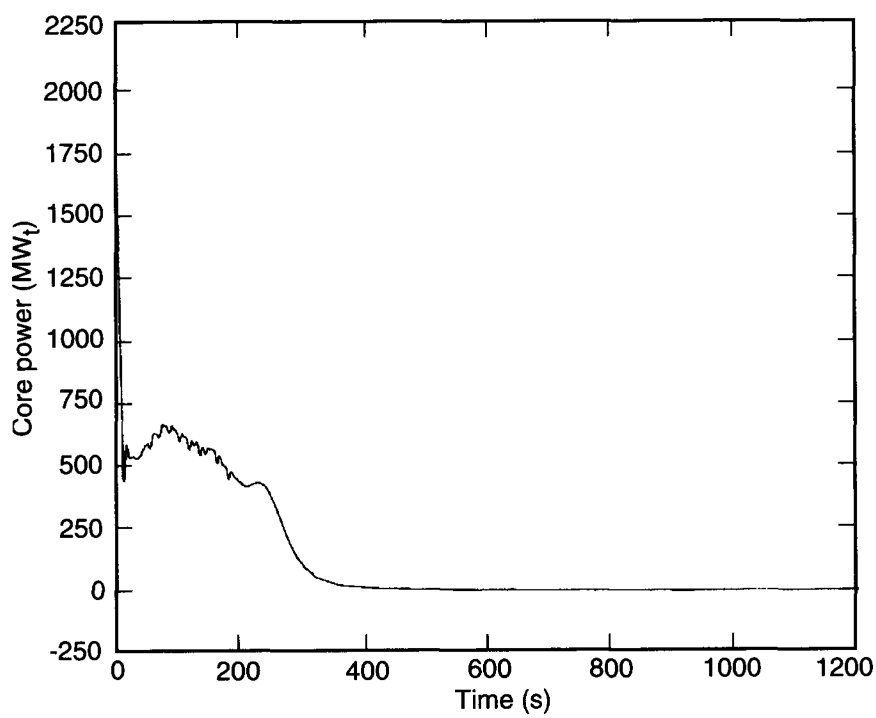

Fig. 11 Core power for an LOSP with complete blockage of the lower density lock. LOSP is loss of off-site power.

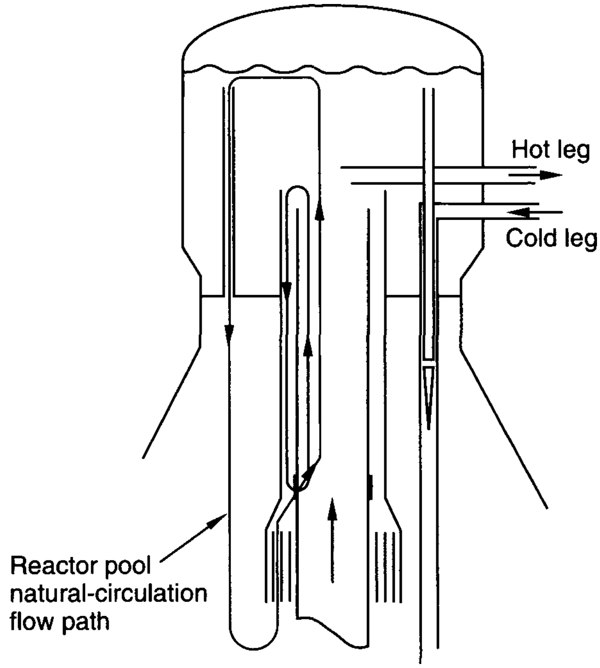

Fig. 12 Natural-circulation flow path for an LOSP with complete blockage of the lower density lock. LOSP is loss of off-site power.

increase only to the concentration of the boron in the pool, which is 1000 ppm. The negative reactivity inserted by the boron is sufficient to produce an initial reduction in core power but is insufficient to produce a reactor shutdown (Fig. 13). The primary pressure begins to increase shortly after the LOSP. The pressure rises to 12.3 MPa, and the safety valves open. The safety valves continue to cycle to the end of the calculated transient at 1200 s. Although a stable condition has been reached, the power level remains high at 500 MW(t); this energy is deposited in the reactor pool. The reactor pool is cooled by both a nonsafety active system and a completely passive safety-grade system. To reach stable decay-heat levels, however, additional boron must be inserted into the primary system.

A test in the ATLE facility simulated an LOSP transient by tripping both of the ATLE recirculation pumps. The key TRAC-calculated result of the assessment calculation, which is the lower density-lock flow, is shown in Fig. 14 along with comparisons to ATLE data and RIGEL-calculated results. The TRAC-calculated peak lower density-lock flow is approximately 25% less than the measured flow. The TRAC-calculated natural-circulation flow rate at the end of the test is approximately 12% less than the measured flow. The RIGEL-calculated peak flow is within 2% of the measured value. The RIGEL-calculated natural-circulation flow rate at the end of the test is approximately 30% greater than the measured flow. Many sensitivity studies were performed to identify the cause of the TRAC underprediction. The lower density-lock flow generally was insensitive to all but one parametric variation. A small increase (15%) in the minimum flow area in the flow path between the riser and the upper density lock led to reasonable agreement with the data (Fig. 14). Because the data used in the TRAC model were scaled from drawings in an area of complex geometry and small dimensions, a 15% error in the flow area is possible. The calculations were repeated with the optional higher order Godunov numeric algorithm activated. For the ATLE LOSP transient, the introduction of solute into the core is rapid, and the reduced numerical diffusion associated with the higher order Godunov method is not significant. Other problems were encountered in modeling the ATLE heat-rod control system; however, the LANL modeling effort was terminated before it was possible to explore these issues fully with ABB. Inadequate knowledge about the facility hardware and operation is thought to be an important contributor to the differences between measured and TRAC-calculated values. Nevertheless, it is evident that the key processes and phenomena of the ATLE test are

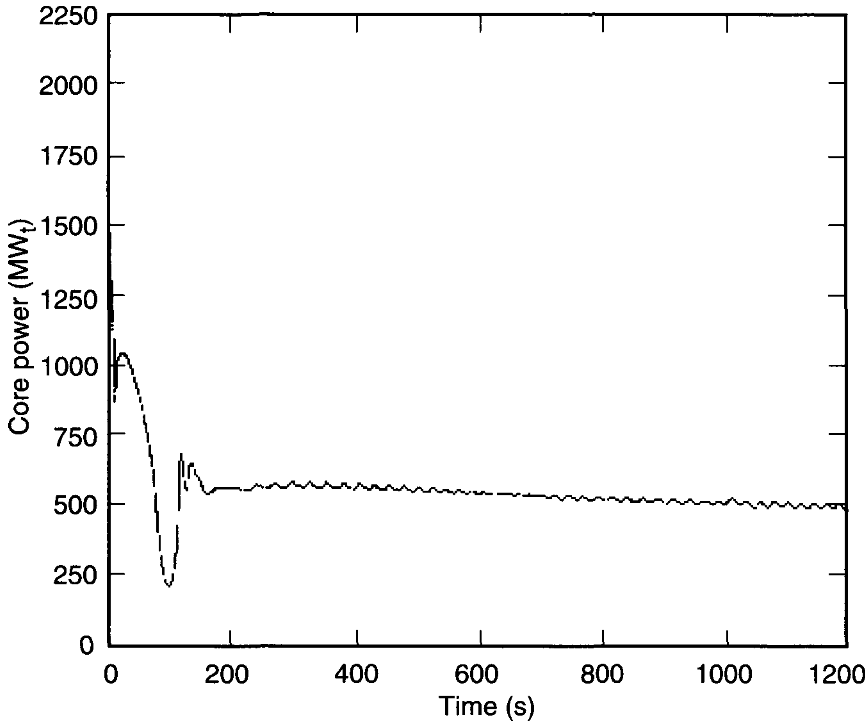

Fig. 13 Core power for an LOSP with a boron concentration of 1000 ppm. LOSP is loss of off-site power.

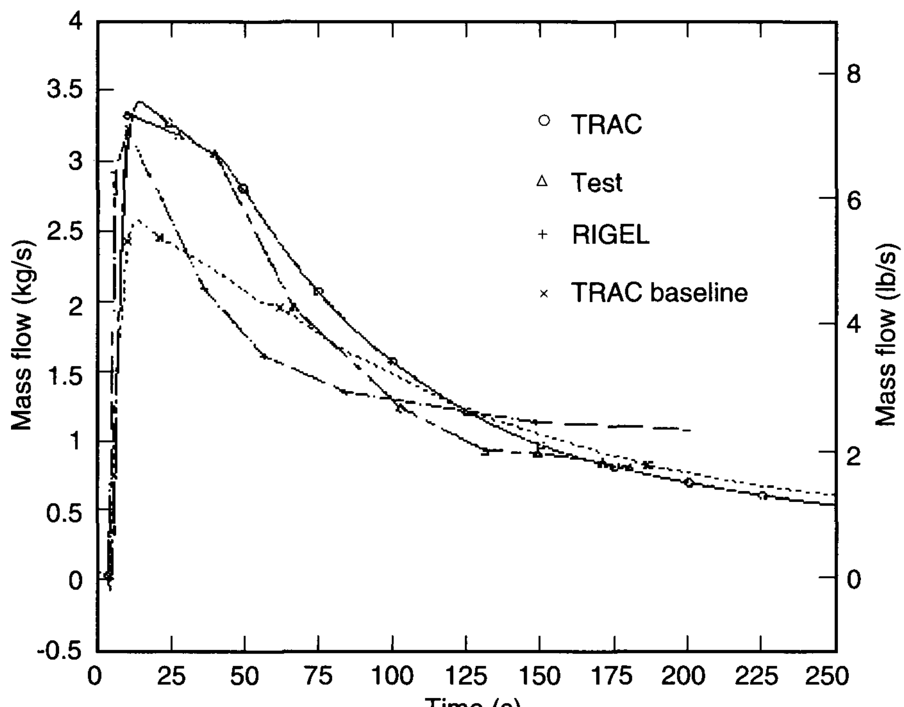

Fig. 14 Comparison of code-calculated and ATLE lower density-lock flows (includes TRAC result for adjusted flow area). ATLE is a 1/308-volume-scale integral test facility that simulates the PIUS reactor.

simulated by TRAC. The underprediction of lower density-lock flow, although of concern, is conservative in that less coolant and negative reactivity from the boron in the pool water is predicted by TRAC. The underprediction influences the early course, but not the final or end state, of a similar transient in the PIUS reactor.

Although a RIGEL calculation of the LOSP for the PIUS supplemental design is not available, the RIGEL simulation of a single RCP trip in the PIUS reactor is available as a benchmark for a code-to-code comparison.[3] The single RCP trip was the programmed trip mode for the original PSID design.[2] The processes and phenomena following a single RCP trip have some similarity to those following an LOSP. The tripped RCP coasts down, whereas the remaining three RCPs increase in speed and rapidly reach their overspeed limit of 115% while attempting to maintain control of the lower density-lock interface. The imbalance caused by the loss of one RCP is, by design, too great for the pump speed control and the lower density locks to activate (Fig. 15). The core power decreases rapidly to shutdown conditions (Fig. 16). The RIGEL-calculated peak lower density-lock flows are higher than those calculated by TRAC. This result is consistent with the results of the ATLE assessment. The RIGEL-calculated power decreases slightly faster than that calculated by TRAC. This trend is consistent with the faster introduction of boron associated with the higher RIGEL-calculated lower density-lock flow. Overall, the early time-calculated results of the two codes are in reasonable agreement. The late time results are in excellent agreement.

MAIN STEAM-LINE BREAK EVENTS

The primary system steady-state boron concentration is 30 ppm, a level characteristic of end-of-cycle operation and the worst-case situation for an MSLB event. The initiating event for the baseline transient is an instantaneous break at the outlet nozzle of a single steam generator. A reactor scram signal is generated when the rapidly decreasing secondary pressure is sensed. The affected steam generator secondary system depressurizes rapidly through the break and thus causes overcooling of the coolant passing through the primary side of the steam generator. The colder liquid from the

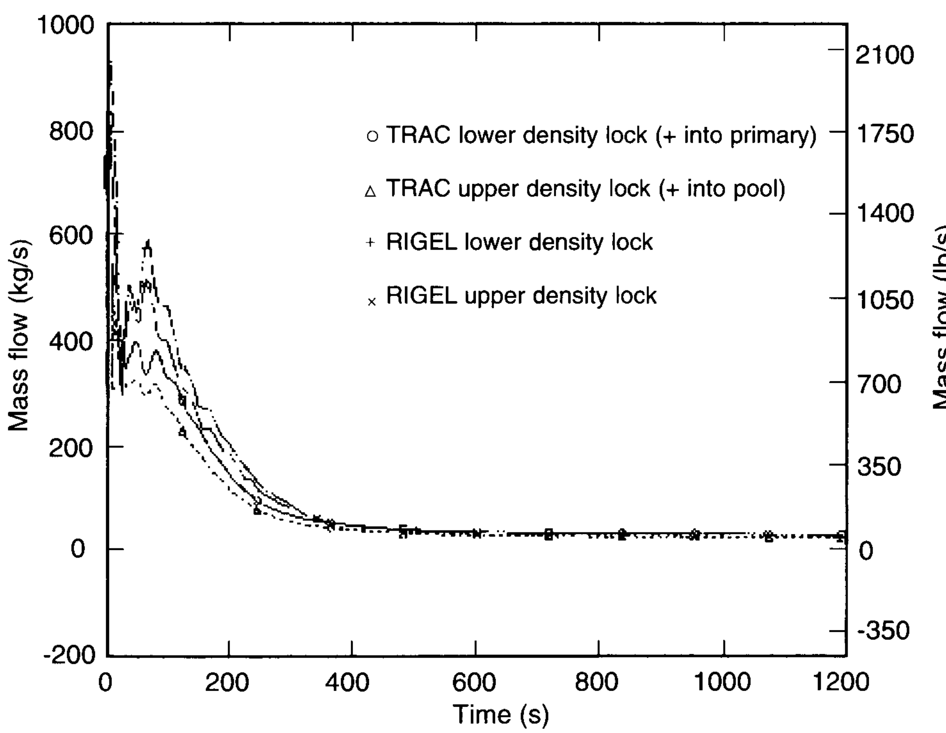

Fig. 15 Lower density-lock flow following loss of a single RCP (co-plots RIGEL and TRAC results). RCP is reactor coolant pump.

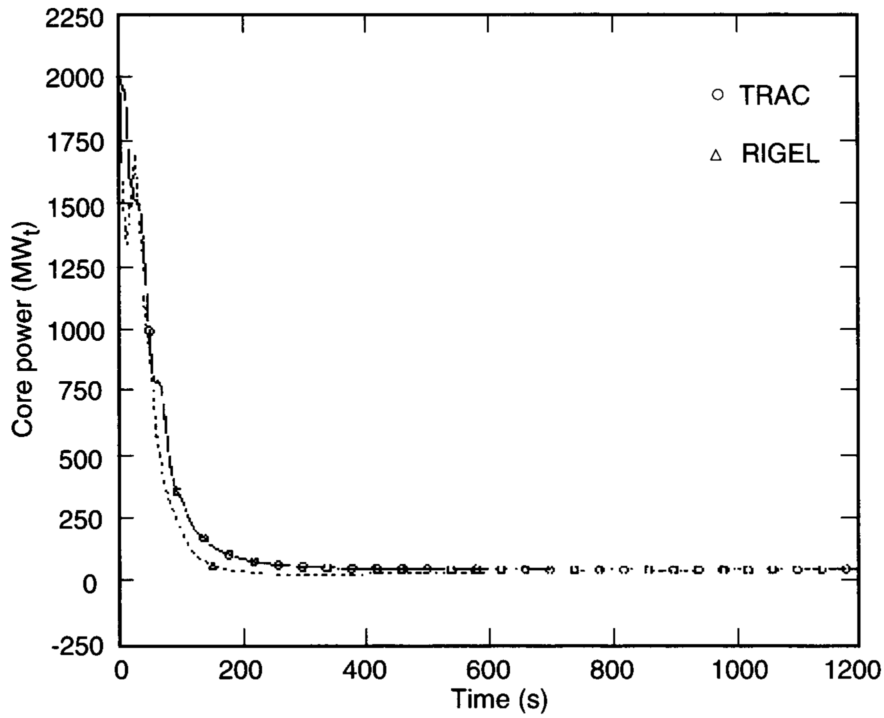

Fig. 16 Core power following loss of a single RCP (co-plots RIGEL and TRAC results). RCP is reactor coolant pump.

overcooled steam generator continues on to the core, where it is a source of positive reactivity. The active-scram system also is initiated early in the transient by the reactor's scram signal. Highly borated water enters the primary system through the scram lines. The increasing core boron concentration is a source of negative reactivity in the core. The total core reactivity, which is the sum of the positive moderator temperature and the negative boron contributions, decreases with the introduction of boron, increases when the cold coolant reaches the core, and then resumes its decrease as highly borated pool water continues to enter the primary system through the scram lines. The core power follows the same trend by decreasing to 1300 MW(t) when the initial boron enters the core, increasing to 1550 MW(t) when the cold coolant enters the core, and finally decreasing to decay-heat levels as highly borated water continues to enter the primary system through the scram lines (Fig. 17). Other than the brief period of positive reactivity resulting from the moderator temperature, the main features of the PIUS primary-system transient are similar to those following an active-scram system transient.

Sensitivity studies were performed to explore the robustness of the PIUS concept when the MSLB is combined with additional low-probability events. The baseline MSLB transient with a concurrent failure of the active-scram system was analyzed. The phenomena occurring in this event sequence differ markedly from the baseline. In the baseline MSLB, positive reactivity inserted by cold water from the affected steam generator is, to a large extent, offset by the negative reactivity inserted by the boron entering the primary system through the scram lines. With the assumed failure of the active-scram system, the early negative reactivity insertion is missing, and the positive reactivity inserted by the coolant is dominant (Fig. 18), which causes the core power to increase to 2550 MW(t) (Fig. 17). Throughout the early transient, control of the lower density-lock interface is maintained so that the highly borated pool remains isolated from the primary system. The primary coolant heats, and the pressure increases to the setpoints of the safety relief valves. These valves continue to cycle for the duration of the calculated transient. Negative reactivity is inserted by primary coolant (moderator) heatup; thus the power

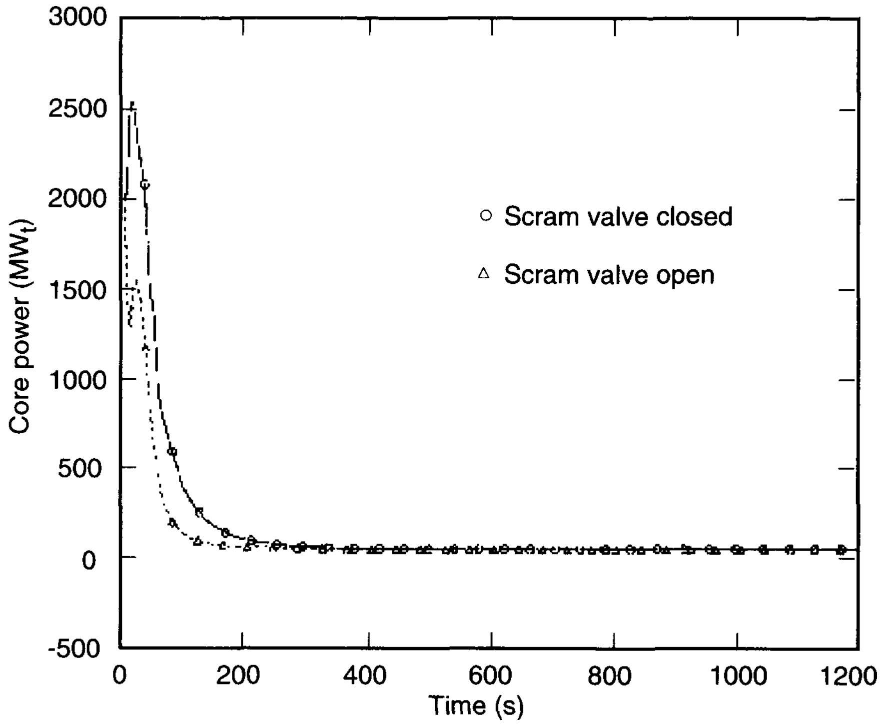

Fig. 17 Core power for MSLB with and without active scram. MSLB is main steam-line break.

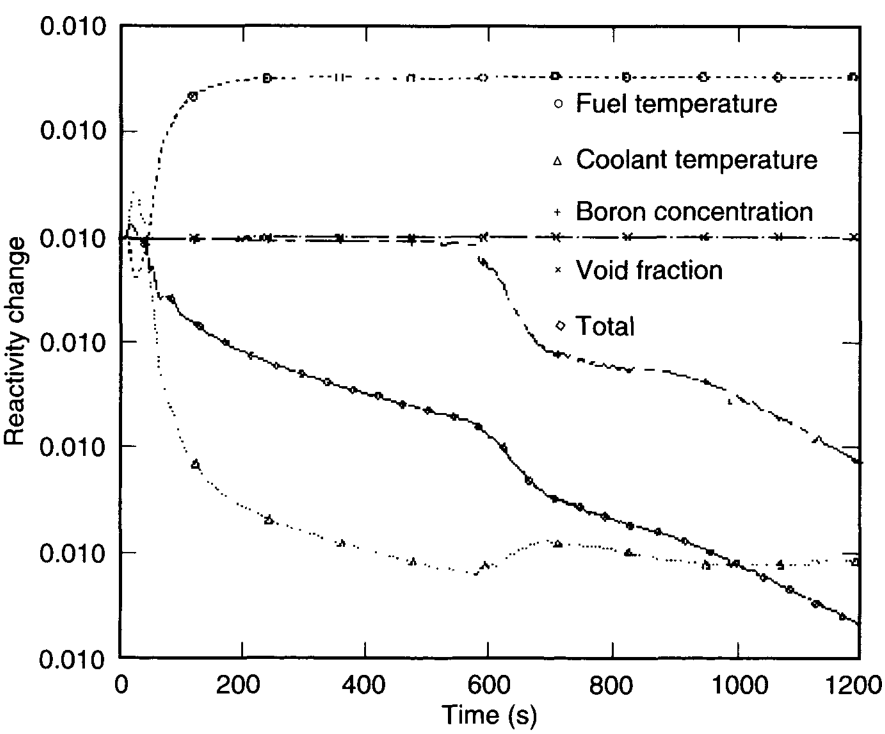

Fig. 18 Core reactivity changes for the MSLB without active scram. MSLB is main steam-line break

begins to decrease. The RCPs are able to maintain control of the lower density-lock interface by increasing speed until the 115% overspeed limit is reached at 520 s. Within 60 s, the lower density lock activates, and a natural-circulation loop is established between the reactor pool and the primary system (as shown by the integrated density-lock flows in Fig. 19). The primary-system boron concentration begins to increase steadily and reaches 160 ppm by the end of the calculated transient. The rate of primary cooldown and depressurization could be increased by tripping one or more RCPs. This transient clearly illustrates the inherent operation of the density locks in the PIUS reactor once the thermal interface in the lower density lock no longer can be maintained. The density locks are activated, and the reactor-pool-to-primary-system natural-circulation loop is established, even though the RCPs continue to operate throughout the calculated transient.

A sensitivity calculation was performed for the baseline active-scram MSLB transient with a concurrent 75% blockage of the lower density lock. The results could not be distinguished from those of the baseline transient because there is little or no flow of highly borated water from the pool to the primary system through the lower density lock during the baseline transient. A final sensitivity calculation was performed for the baseline MSLB transient with a concurrent boron concentration of 1800 ppm. Although there were slight differences in the course of the calculated transients, the differences were not significant. The reduction of the core power to decay levels was delayed slightly by the lower concentration of boron entering the primary system from the pool. After the scram valves were closed, the primary boron concentration stabilized at 500 ppm, as compared with 600 ppm in the baseline. This led to slightly elevated coolant temperatures throughout the transient. Although not calculated, the response of PIUS to an MSLB baseline transient with a concurrent boron concentration of 1000 ppm is expected to be similar to that previously described for the active scram with a boron concentration of 1000 ppm.

SBLOCA EVENTS

The initiating event for the baseline transient is a break in the pressure relief system piping at the flange just outside the steel pressure vessel and upstream of the safety relief valves (Fig. 1). Steam flows through the break at a peak rate of 105 kg/s and then decreases in

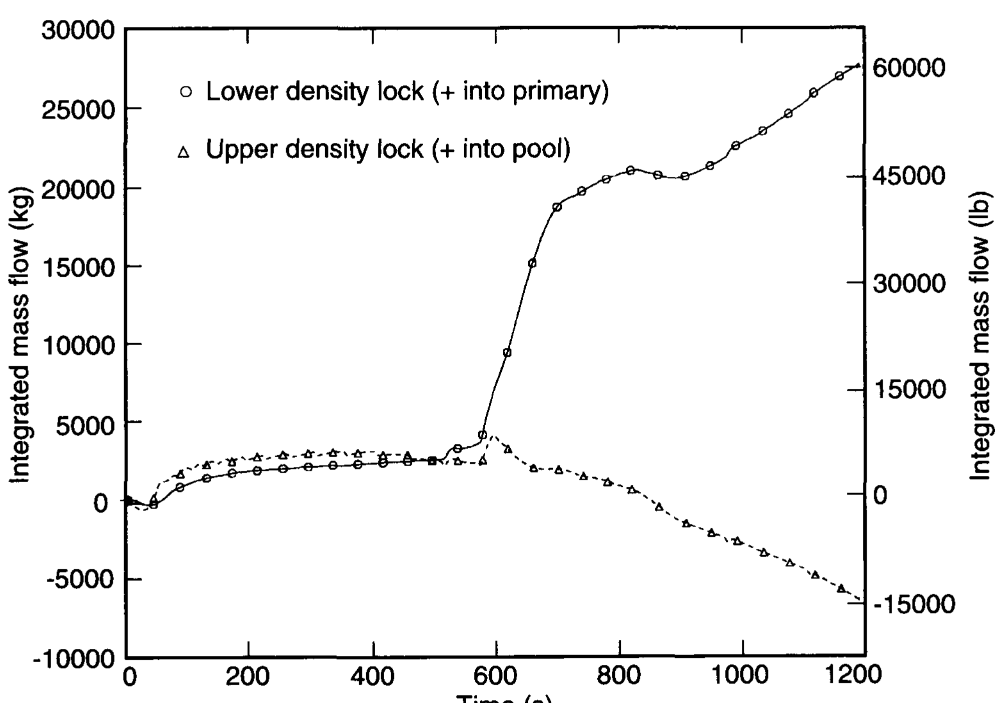

Fig. 19 Integrated density-lock flows for the MSLB without active scram. MSLB is main steam-line break.

concert with the primary pressure until a two-phase flow through the break begins at 230 s. At that time the flow rate increases temporarily to 110 kg/s and then resumes its decrease to 50 kg/s at 1200 s. A scram is initiated at 18 s when the primary system depressurizes to 8.5 MPa. Injection of highly borated water into the primary system through the scram lines causes a rapid decrease in the core power to decay levels. As previously described for active-scram events, during the interval that the scram valves are open, inventory is displaced from the primary system into the reactor pool, primarily through the upper density lock. While the scram valves are open, the RCP inlets are full of liquid; however, closure of the scram valves at 230 s induces a marked change in the primary system behavior. Immediately following termination of the scram-line flow, voiding occurs in the pump inlets (Fig. 20), the RCPs increase to their overspeed limit of 115% of nominal (Fig. 21), and, subsequently, the RCP discharges become oscillatory. The oscillatory behavior of the RCP discharges propagates throughout the primary system [for example, the density-lock flows are highly oscillatory (i.e., flow oscillations of ±600 kg/s and a frequency of 25 s)]; however, a net circulation pattern is established with pool water entering the primary system through the lower density lock and exiting the primary system through the upper density lock (Fig. 22). The net inflow through the lower density lock produces a continuing, albeit oscillatory, increase in the primary boron concentration. For the most part, coolant temperatures decrease throughout the transient; however, the core inlet temperature increases following closure of the scram lines, and the core outlet periodically saturates as the core flow oscillates in concert with the RCP discharges.

A RIGEL calculation of the first 300 s of an SBLOCA in the pressure relief system piping has been reported.[3] The TRAC and RIGEL results generally are in qualitative agreement until 230 s, when the scram valves close. There are moderate differences in the parameter values; however, the same trends are predicted by the two codes. There are important phenomenological differences between the two calculations after 230 s; however, these differences are believed to arise from the timing at which events occur and, when considered in the perspective of extended transient times (e.g., 1200 s), are not significant. The TRAC-calculated results show that the RCP controller

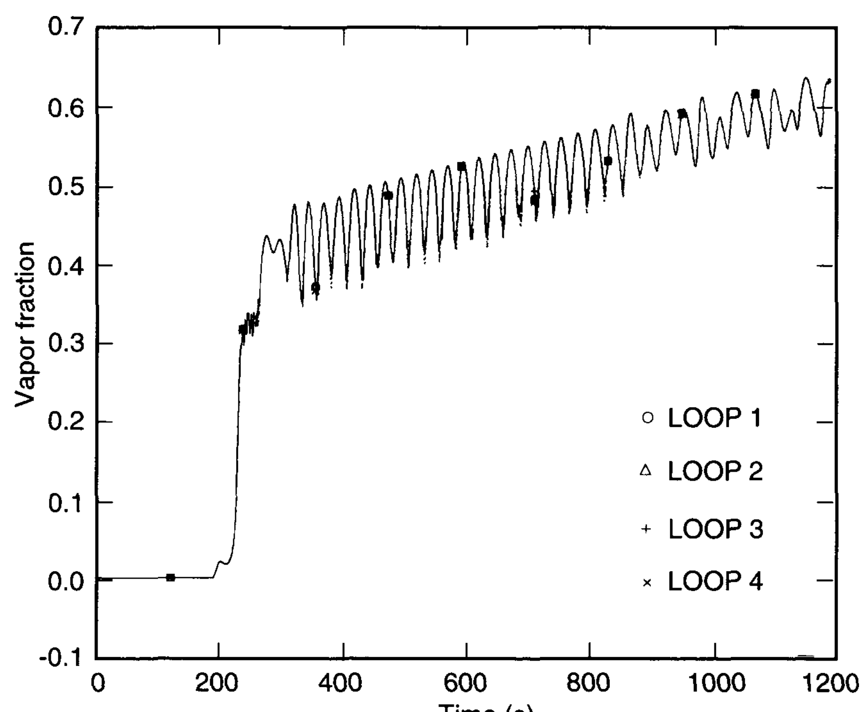

Fig. 20 Void fraction in the RCP inlets for the primary-relief-line SBLOCA baseline case. RCP is reactor coolant pump; SBLOCA is small-break loss-of-coolant accident.

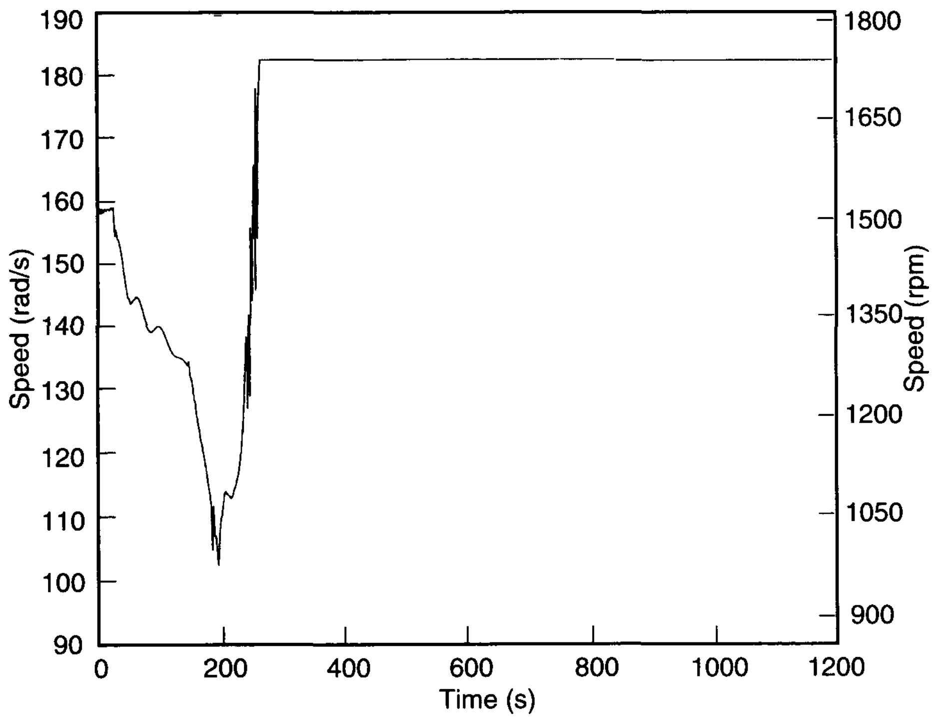

Fig. 21 RCP pump speeds for the primary-relief-line SBLOCA baseline case. RCP is reactor coolant pump; SBLOCA is small-break loss-of-coolant accident.

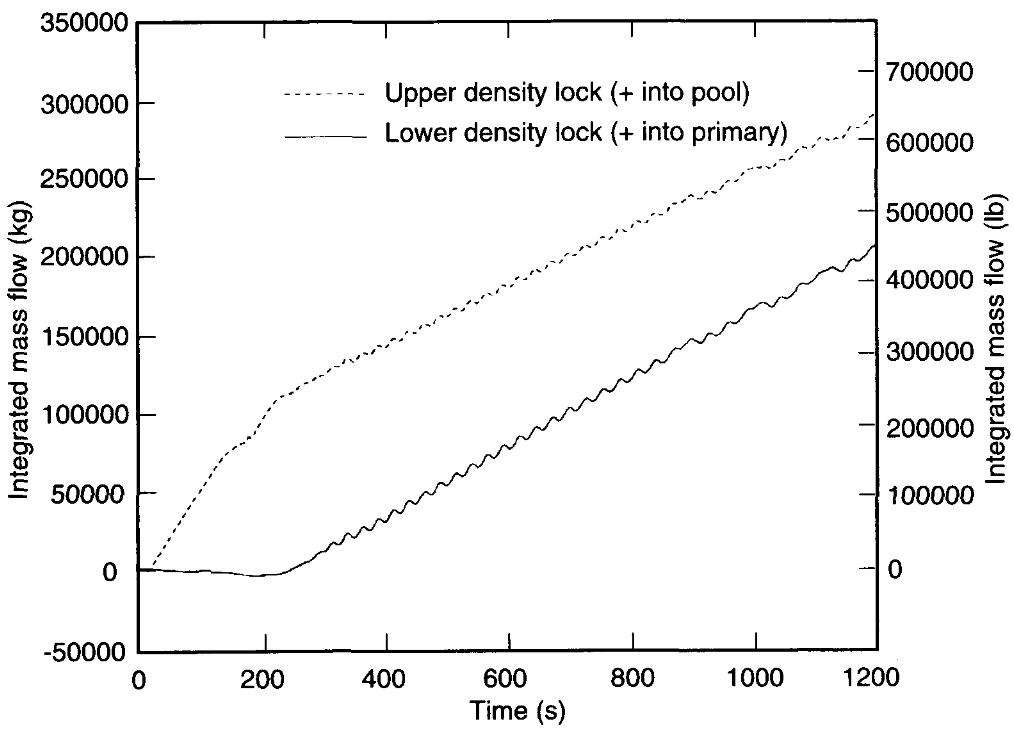

Fig. 22 Integrated density-lock flows for the primary-relief-line SBLOCA baseline case. SBLOCA is small-break loss-of-coolant accident.

demands an increase in speed at 210 s, which is approximately 10 s after the scram valves begin to close. The RCP overspeed limit is reached by 260 s. The flow oscillations predicted by TRAC arise approximately 40 s after the RCPs have reached their overspeed limit and are caused by voiding in the RCP inlets subsequent to closure of the scram valves. The RIGEL-calculated results show that the RCP controller demands an increase in speed at 255 s and that the 115% overspeed limit is reached shortly before 300 s. The authors believe that oscillatory RCP flows would be calculated by RIGEL for times that are greater than 300 s. A RIGEL calculation was reported for a break in the same location for the original PSID design.[2] During that transient the RCP outlet flows were oscillatory after voiding arose in the inlets to the operating RCPs and after the RCP overspeed limit was reached.

Sensitivity studies were performed to explore the robustness of the PIUS concept when blockage to the lower density lock occurs. The first study examined the response of the PIUS reactor to the baseline SBLOCA, concurrent with a 75% blockage of the lower density lock. The baseline and 75% blockage results are similar in all major trends and average quantities; however, there is an important phenomenological difference between the two calculations. The baseline calculation displays a strong oscillatory character when the RCP inlets void following termination of the scram-line flows. The blockage case is markedly different. Oscillations during the few intervals of existence are much smaller and decay with time. Partial blockage of the lower density lock appears to "stiffen" the combined primary system–pool system, which results in pump-induced oscillations that do not grow to detectable levels and, when they do become detectable, are damped. The second sensitivity study examined the response of the PIUS reactor to the baseline SBLOCA, concurrent with a boron concentration in the pool of 1800 ppm. The lowered boron concentration was of no consequence; the only impact was that the time lengthened slightly to reduce primary-system temperatures to the same level as those which occurred in the baseline. Oscillatory behavior occurred in this sensitivity calculation. The third sensitivity study examined the response of the PIUS reactor to the baseline SBLOCA, concurrent with a failure of the active-scram system. Similar end states were reached for the two calculations by 1200 s when the transient calculations were terminated. The course of the sensitivity study transient differed in several respects, however. The core power decreased more slowly than in the baseline because there was no rapid injection of boron from the active-scram system. The initial decline in core power was caused by the negative reactivity insertions from increasing moderator temperatures and voiding in contrast to the baseline, where the only source of negative reactivity was from boron entering the core. Oscillatory primary-system behavior was calculated.

As a final sensitivity study, a small break in a second location was analyzed—a break in a single scram line at a location near the RCP inlet. The diameter of the scramline is slightly less than twice that of the primary relief line, and coolant is lost from both the pool and pump ends of the break; however, flow through the pump side of the break does not start until the scram valve opens. Coolant is lost only through one end of the primary relief-line break. Thus the scram-line SBLOCA is a more severe accident, as measured by the amount of coolant lost from the system. The larger primary inventory loss affects operation of the active-scram system. The pool-side and pump-side break flows are shown in Fig. 23. Both break flows rapidly decrease from the maximum levels reached immediately following break initiation. The decreasing break flows are the result of a rapidly falling primary-system pressure and voiding at the break inlets. Early in the transient, the primary source of negative reactivity is from boron injected by the active-scram system following system activation on a low primary-pressure signal. The active-scram system is effective only for the first 40 s of the transient because flows through the intact scram lines decrease rapidly when the pool liquid level approaches the elevation of the scram-line nozzle connections. The negative reactivity inserted during the period of active-scram-system operation rapidly reduces the core power to 1250 MW(t) (Fig. 24). The RCPs maintain control of the lower density-lock interface until approximately 55 s. Between 40 and 55 s, a power-to-flow mismatch exists, with power near the 1250-MW(t) plateau and core flow decreasing. The coolant (moderator) temperature increases and partial voiding of the core occurs, both of which cause negative reactivity to further decrease the power to 380 MW(t) by 55 s. At this time the RCPs reach their overspeed limit of 115%, the lower density lock activates, highly borated pool water enters the core, and the core power decreases to shutdown levels. For much of the transient, the flows through the upper and lower density locks are highly agitated; however, the integrated density-lock flows clearly show a net natural circulation from the pool into the primary system through the lower

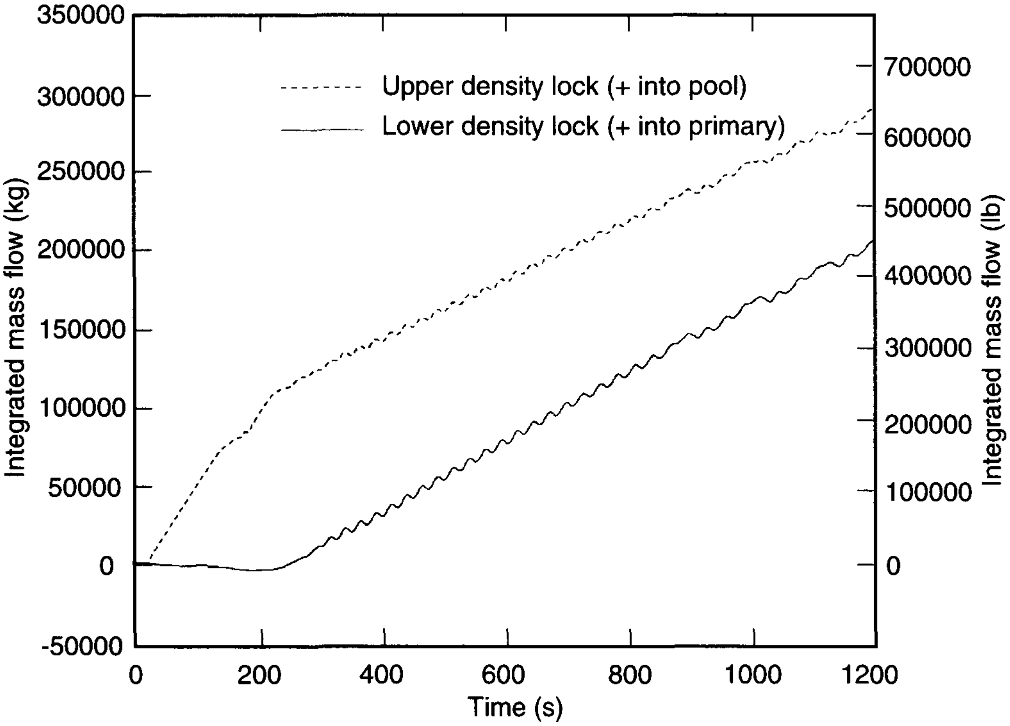

Fig. 23 Break flows for the scram-line SBLOCA case. SBLOCA is small-break loss-of-coolant accident.

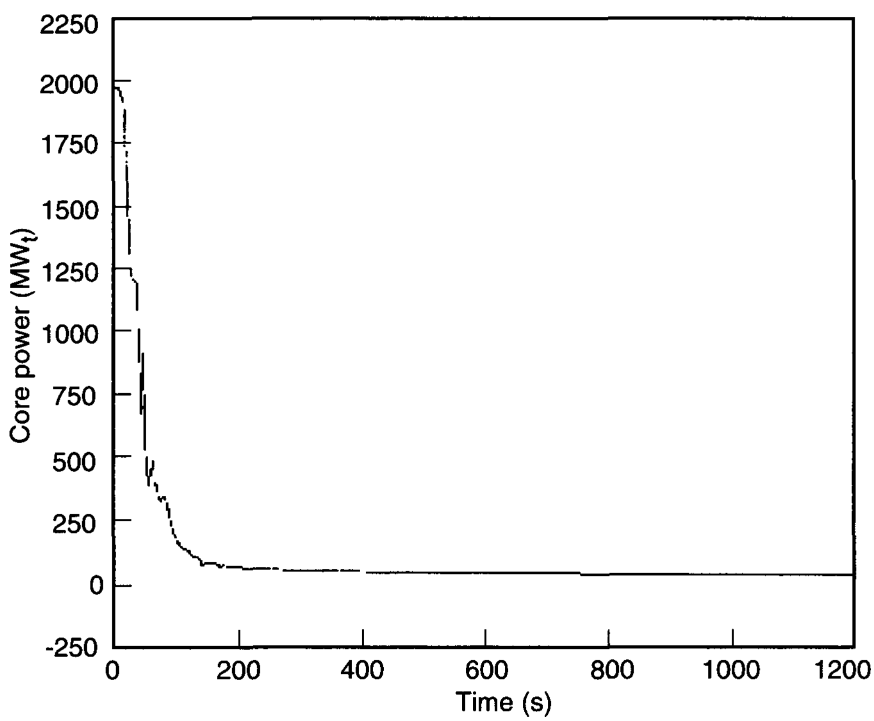

Fig. 24 Core power for the scram-line SBLOCA case. SBLOCA is small-break loss-of-coolant accident.

density lock, and a return flow to the pool via the upper density lock is established (Fig. 25). Thus, by 1200 s, the loss of coolant through the ends of the scram-line break is negligible, the core power is at shutdown levels, the loops are voided, and a natural-circulation flow between the primary system and pool through the density locks is fully established.

LBLOCA EVENTS

The initiating event for the baseline transient is a double-ended guillotine break in one cold leg just outside the steel pressure vessel. A scram is initiated at approximately 1.5 s, when the primary system depressurizes to 8.5 MPa. The break flows from the vessel side and RCP side of the break are shown in Fig. 26. Both flows decline rapidly as the primary-system pressure decreases and voiding in the break flows increases. The active-scram system injects borated water only for the first 11 s of the transient, while the reactor pool level is above the scram-line nozzle connections to the pool. In addition, most of the pool water injected through the scram lines is discharged out the break. Immediately after the start of the LBLOCA, flows in both the core and downcomer reverse (Fig. 27). The flow reversal lasts approximately 8 s; during this period a large fraction of the core reaches saturation temperatures and voids (Fig. 28). The initial period of core voiding is terminated when the downcomer and core flows resume their normal flow direction and coolant reenters the core from the lower plenum. This occurs when flows from the intact cold legs enter the cold-leg plenum, and flows to the break can supply the rapidly decreasing vessel-side break flow fully. Before that time, vessel inventory, as well as flows from the intact loops, is needed to supply the break flow.

A second core-flow reversal begins at approximately 20 s and continues until 30 s. Before this time the inlets of the RCPs begin to void, and RCP performance degrades. With the sharp decrease in pumped flow, saturation temperatures again are attained in much of the core; the resultant void generation causes the core flow to reverse. The reverse core flow peaks at 25 s, when hot fluid from the riser enters the core from the top, vaporizes in the core, and reduces the downward mass flow at the core inlet. At approximately 30 s the voids in the core collapse, and thus lower-plenum fluid

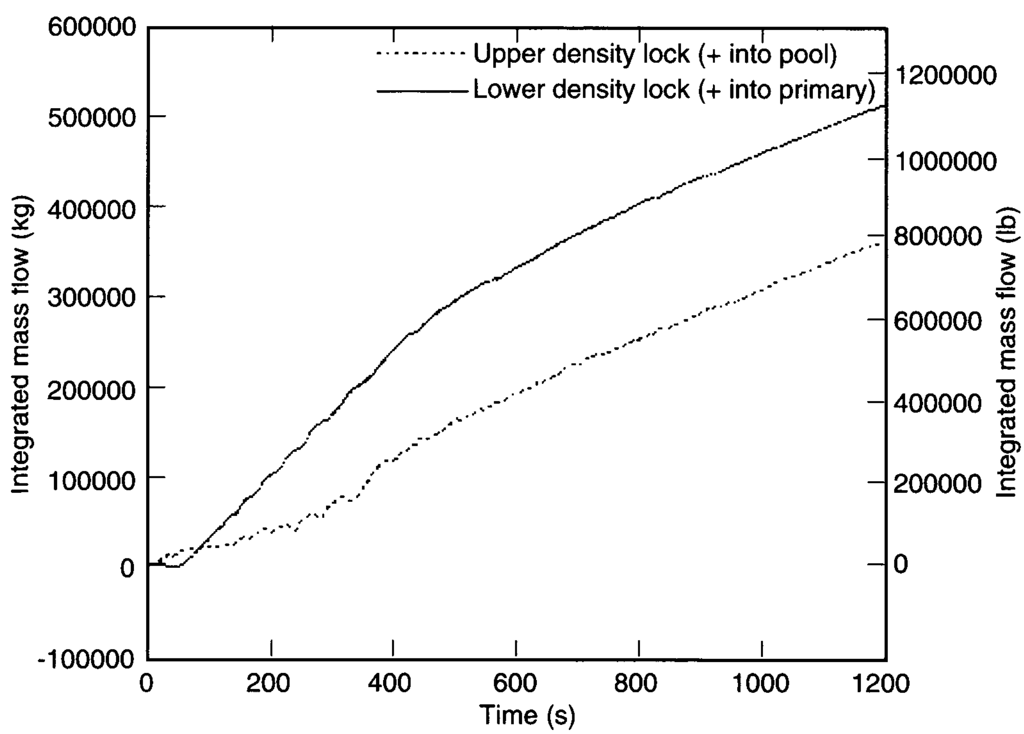

Fig. 25 Integrated density-lock flows for the scram-line SBLOCA case. SBLOCA is small-break loss-of-coolant accident.

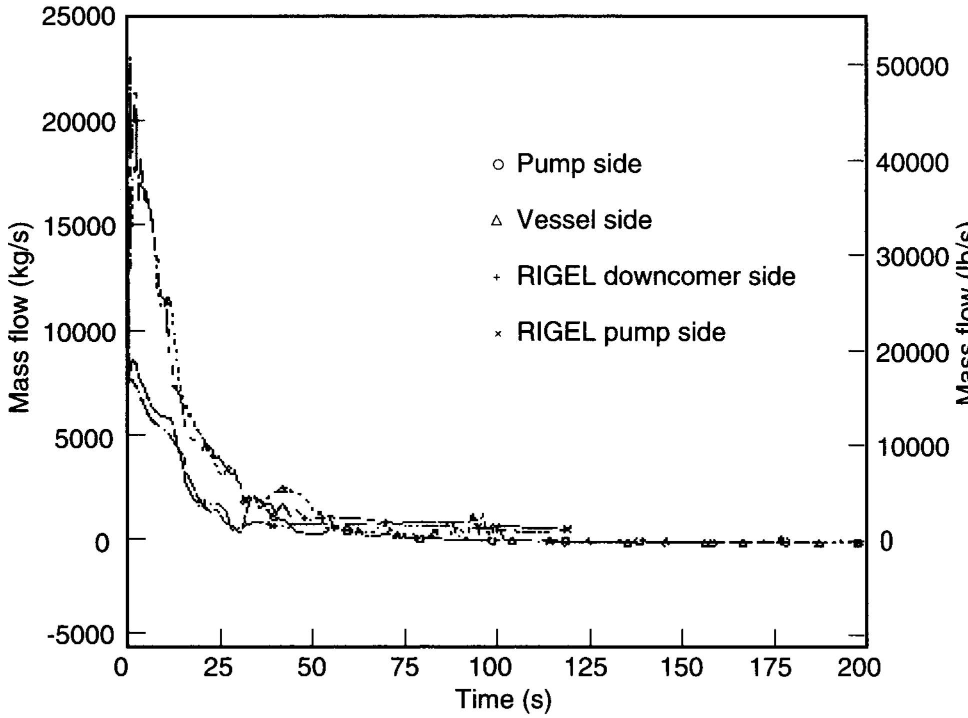

Fig. 26 Break flows for the 1-D LBLOCA baseline case. LBLOCA is large-break loss-of-coolant accident.

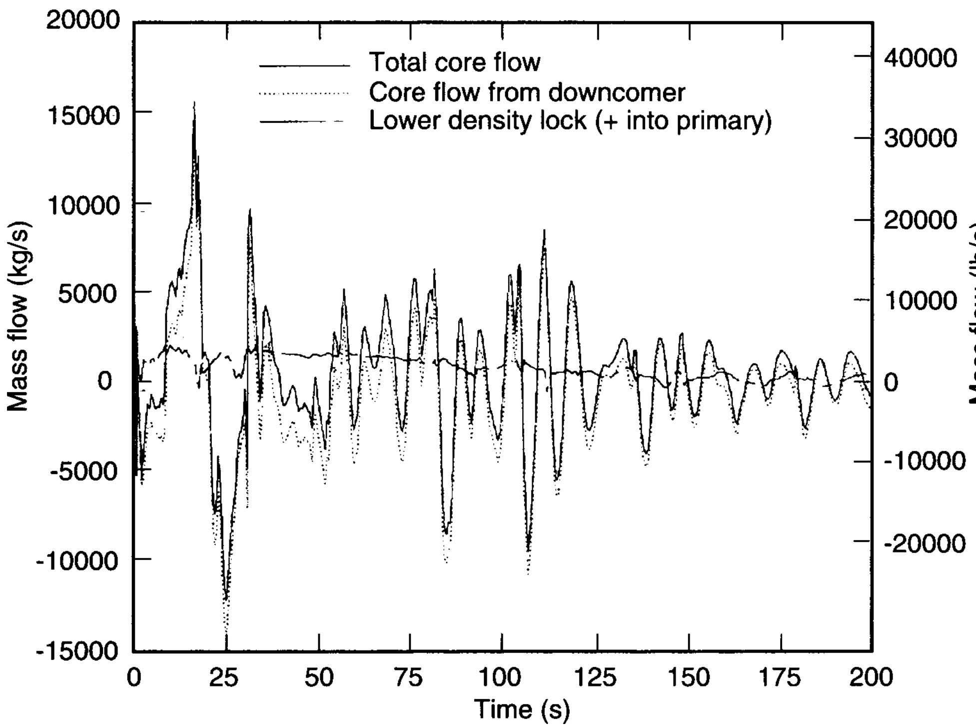

Fig. 27 Core, downcomer, and lower density-lock flows for the 1-D LBLOCA baseline case. LBLOCA is large-break loss-of-coolant accident.

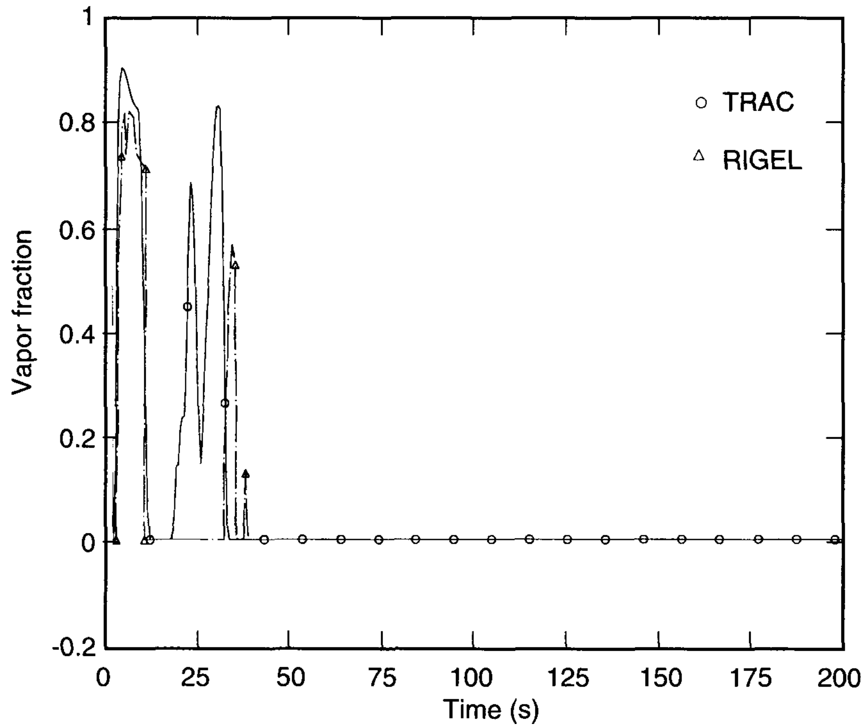

Fig. 28 Core average void fraction for the 1-D LBLOCA baseline case. LBLOCA is large-break loss-of-coolant accident.

surges briefly into the bottom of the core. After 30 s the core remains liquid full, and by 60 s recovery of the primary inventory (refilling the riser above the core) is under way (Fig. 29). The decay heat is removed by the break flow and by the natural circulation of pool water that enters the primary system through the lower density lock, passes through the core, and reenters the pool through the upper density lock.

The core power rapidly decreases immediately following the LBLOCA initiation (Fig. 30). Voiding in the core is the single largest negative reactivity component early in the transient (Fig. 31). There is a sharp 2-s rise in core power to 1150 MW(t), beginning at 15 s. The core power subsequently decreases to decay levels and remains there for the rest of the calculated transient. The brief period of criticality that begins at 15 s occurs as the core refills after the first flow reversal. The negative void reactivity is eliminated, and positive reactivity is inserted as primary coolant and pool water reenter the core. Although the pool water is highly borated and inserts negative reactivity, the primary coolant inserts positive reactivity because it reduces the fluid temperature of the core. The net result is a brief interval when the core is critical. Neither the core dryout nor cladding temperature heat-up excursions are calculated (Fig. 32) during the transient.

A RIGEL calculation of this LBLOCA has been reported.[15] In general, the TRAC- and RIGEL-calculated results display the same phenomena and trends; however, there are differences in the details. The calculated break flows are compared in Fig. 26. The RCP-side break flows are similar. The RIGEL-calculated peak vessel-side break flow is approximately 23 000 kg/s, whereas the TRAC-calculated maximum flow is 17 800 kg/s. This result suggests that there may be differences between the RIGEL and TRAC critical flow models. The immediate reversal of the downcomer and core flows and the complete bypass of the lower density-lock flow are predicted by both codes; however, the magnitude of the RIGEL-calculated peak-reversed core flow is greater than that calculated by TRAC, and the flows are approximately 10 000 and 3 700 kg/s, respectively. This result is consistent with the peak vessel-side break flow calculated by RIGEL, which is approximately 5 200 kg/s larger than that calculated by TRAC. The RIGEL-calculated core-flow reversal lasts until nearly 10 s, whereas the TRAC-calculated flow reversal ends shortly after 8 s. Because

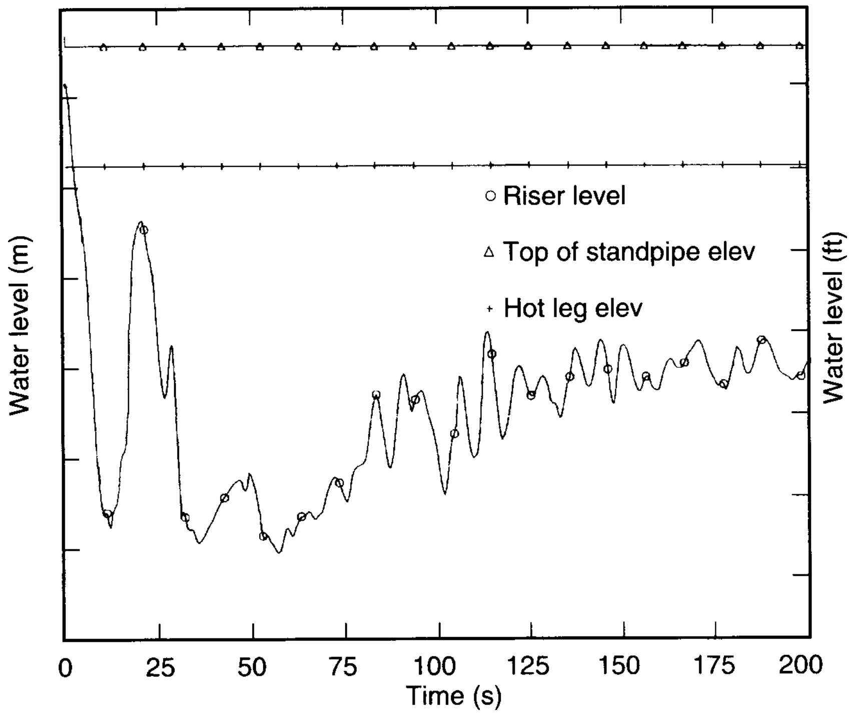

Fig. 29 Collapsed liquid level for the 1-D LBLOCA baseline case. LBLOCA is large-break loss-of-coolant accident.

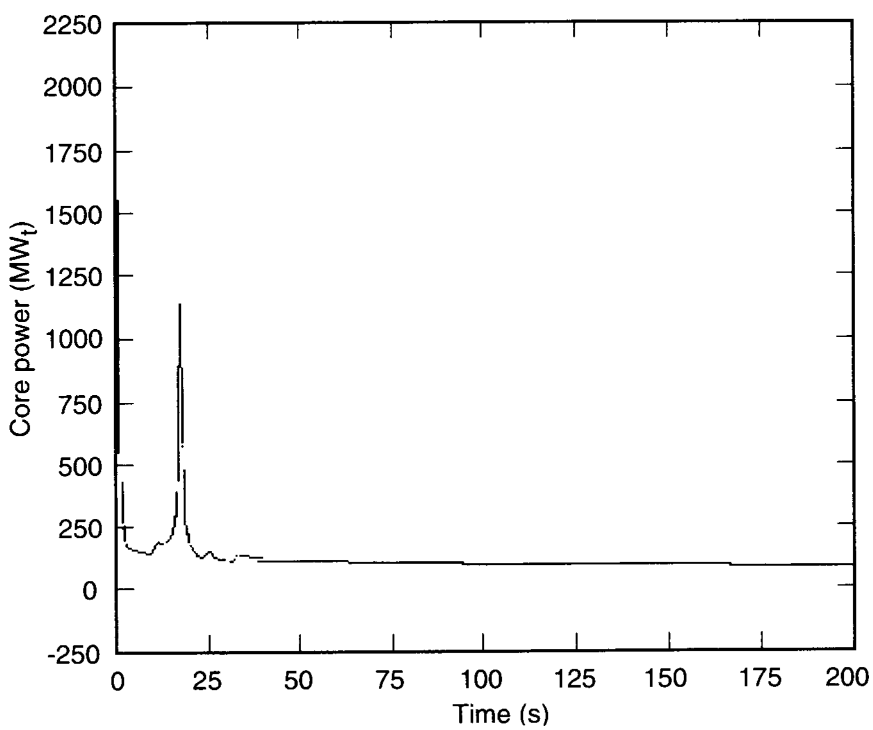

Fig. 30 Core power for the 1-D LBLOCA baseline case. LBLOCA is large-break loss-of-coolant accident.

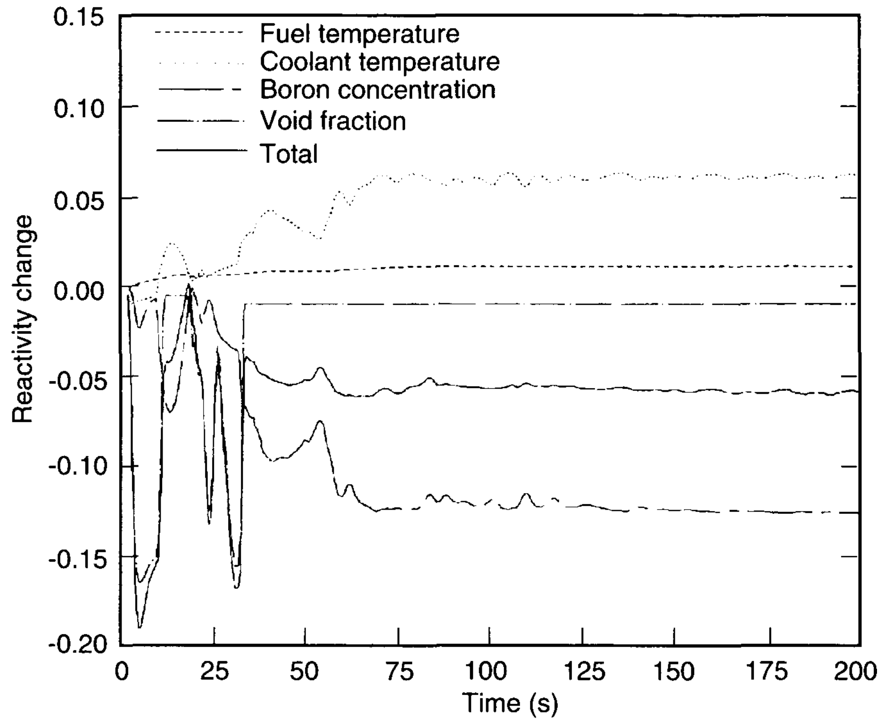

Fig. 31 Core reactivity changes for the 1-D LBLOCA baseline case. LBLOCA is large-break loss-of-coolant accident.

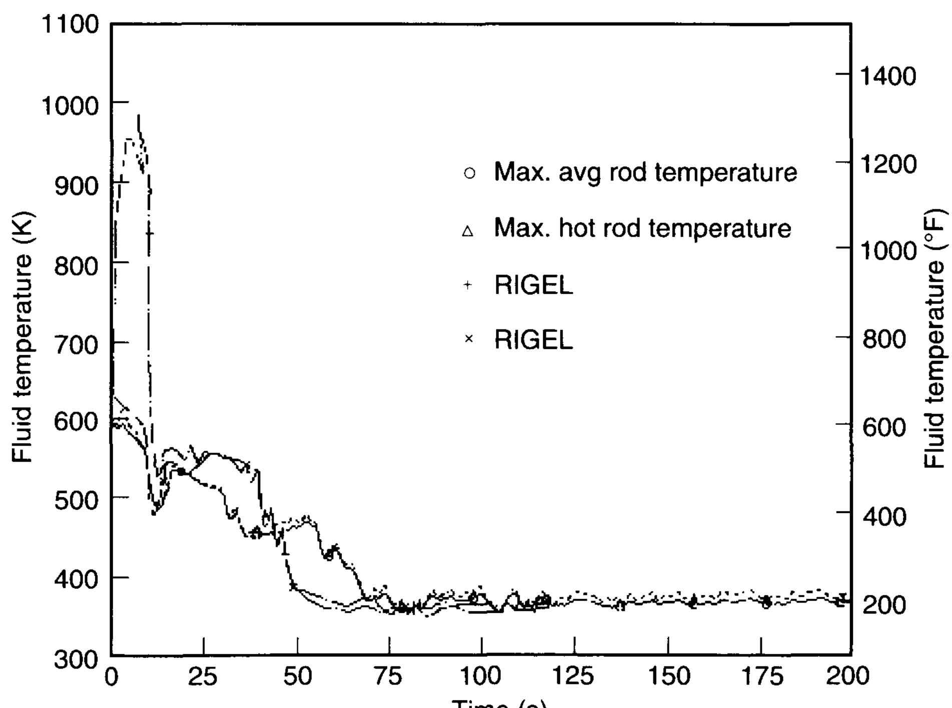

Fig. 32 Cladding temperatures for the 1-D LBLOCA baseline case. LBLOCA is large-break loss-of-coolant accident.

the flow reversal predicted by RIGEL lasts longer, the period of voiding in the core also is extended. Consequently RIGEL calculates a dryout and heatup of the hot rod in the model to approximately 990 K (1 323 °F), which may be compared to the 10 CFR 50.46 licensing limit of 2 200 °F (Fig. 32); however, the calculated uncertainty in the calculated peak cladding temperature has not been quantified. The later termination of the initial flow reversal in the RIGEL calculation is consistent with the understanding of why the flow reversal terminates (i.e., that the break flow has decreased to the point that the break can be supplied by the intact loop cold-leg flows). It is clear that the magnitude of the vessel-side break markedly affects the early details of the predicted LBLOCA transient. In summary, both TRAC and RIGEL predicted the same major phenomena and processes, and both predict that the reactor will reach shutdown conditions without damage. There are important differences in details, however, particularly with respect to the magnitude of the vessel-side break flow; these differences influence the predicted courses of the LBLOCA transient.

Several sensitivity studies were completed with the 1-D model. The first study examined the response of the PIUS reactor to the baseline LBLOCA concurrent with a 75% blockage of the lower density lock. The phenomena occurring during this low-probability transient were similar to the baseline. The same core-flow reversal pattern occurred for the same reasons that were discussed previously. During periods of positive core flow, however, the flow rates through the core were smaller because the flow entering the primary system through the lower density lock was reduced by the lower density-lock flow blockage. The amount of boron entering the core through the lower density lock also was reduced. Voiding in the core was greater during the second and third core-flow reversal periods. Thus, during the calculated transient, voiding contributed more to the total negative core reactivity and boron contributed less. After the initial decrease in core power and immediately following LBLOCA initiation, a power increase again was calculated. The power increased to approximately 1 100 MW(t), which is less than in the baseline. Neither cladding dryout nor cladding heatup was predicted. The second sensitivity study examined the response of the PIUS reactor to the baseline LBLOCA concurrent with a boron concentration in the pool of 1 800 ppm. The course of this transient was nearly identical to the baseline with one exception. The core power increase beginning at approximately 15 s is more severe than in the baseline because there is less negative reactivity inserted in the core by the pool coolant; however, there is no core dryout or heatup. The third sensitivity study examined the response of the PIUS reactor to the baseline LBLOCA concurrent with a failure of the active-scram system. The impact was minimal. For the baseline transient, the active-scram system is effective only for the first 11 s of the transient, after which the reactor pool level drops below the level of the scram-line takeoff from the pool. Because the core flow is reversed for the first 6.5 s of the transient, the active-scram system has a limited impact on the course of the baseline transient. Thus the course of the transient for the sensitivity calculation was nearly identical to the baseline calculation.

The second baseline LBLOCA calculation was performed with the 3-D input model. Because a combined multidimensional neutronics and thermal–hydraulic modeling capability was lacking, only a few 3-D calculations were performed. In major phenomena and trends, the 1-D and 3-D calculations are similar, although there are some differences in detail. There are no differences that can be attributed specifically to the multidimensional model. The calculated peak break flows for the 1-D and 3-D baseline transients are similar; however, the vessel-side break flow remains higher in the 3-D calculation after the transition to a two-phase break flow at 18 s. The higher vessel-side break flow results in a faster depressurization in the 3-D calculation. The core power exhibits an early decrease to decay-heat levels followed by a subsequent power increase to approximately 920 MW(t) at approximately 18 s. The predicted core power increase is somewhat less than the approximate 1 150 MW(t) peak calculated for the 1-D baseline calculation and occurs approximately 3 s later. The initial core-flow reversal lasts approximately 7 s and is terminated when the vessel-side break flow can be supplied by the coolant flow through the intact loops. The subsequent positive core flow is terminated when the inlets of the RCPs void and pump performance degrades. A second period of reverse core flow then occurs that terminates at the end of the power increase as voids collapse in the core. These phenomena are the same as those in the 1-D baseline. The following differences are noted. A third period of reverse core flow occurs in the 3-D calculation and thus causes voiding in the core from 55 to 62 s. Core voiding also is predicted from 78 to 110 s because of the lower system pressure in the 3-D calculation. In general, the differences between the 1-D and 3-D calculations do not appear to be significant.

SUMMARY OBSERVATIONS

1. Reactor shutdown to decay-heat levels is predicted for each of the five baseline initiating events. The active-scram system effectively reduces core power to decay levels for reactor scram, MSLB, and SBLOCA events. The passive-scram system effectively reduces core power to decay levels for transients in which the scram system is either unavailable (e.g., LOSP events) or inoperable (e.g., LBLOCA events after the pool water level declines below the scram-line nozzles).

2. As presently conceived, the PIUS core has inherent, compensating neutronic shutdown mechanisms. PIUS also has multiple flow paths between the primary system and reactor pool. Alternate flow paths exist, even if complete blockage of either density lock occurs. Neither operator nor active system actions are needed to accomplish reactor shutdown, even for the various event initiators combined with very-low-probability occurrences.

3. Confidence in the baseline simulations is enhanced by the assessment activity performed with the use of ATLE data. The ATLE processes and phenomena were predicted correctly by TRAC. Quantitative discrepancies occur between key TRAC-calculated parameter values and the ATLE data; the reasons for these differences should be understood if PIUS is submitted to the NRC for design certification.

4. Our confidence in the predicted outcomes of the baseline simulations is enhanced by the code-to-code benchmark comparisons that have been conducted for the active-system scram, SBLOCA, and LBLOCA. RIGEL and TRAC are two independently developed codes, yet the RIGEL- and TRAC-calculated results display many areas of similarity and agreement. There are also differences in the details of the transients and accidents calculated by the two codes. The reasons for these differences should be explored if PIUS is submitted to the NRC for design certification.

5. Although the sensitivity calculations performed for each event type explore sequences well beyond the base of code assessment and code-to-code benchmark evaluations, the analyses reported here indicate that PIUS will accommodate such low-probability sequences successfully. No phenomenological "cliffs" were encountered in any of the sensitivity studies.

6. Combined multidimensional core neutronic and thermal–hydraulic effects are thought to be important and should be investigated thoroughly if the design and safety review effort continues.

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The Hierarchy-By-Interval Approach to Identifying Important Models that Need Improvement in Severe-Accident Simulation Codes

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R. P. Jenks-Johnson,[d] and Y.-S. Chen[e]

Abstract: *The hierarchy-by-interval (HBI) methodology was developed to determine an appropriate phenomena identification and ranking table for an independent peer review of severe-accident computer codes. The methodology is described, and the results of a specific code review are presented. Use of this systematic and structured approach ensures that important code models that need improvement are identified and prioritized, which allows code sponsors to more effectively direct limited resources in future code development. In addition, critical phenomenological areas that need more fundamental work, such as experimentation, are identified.*

An independent computer code peer review process recently developed to assist the U.S. Nuclear Regulatory Commission (NRC) in their nuclear safety missions[1–3] has been used to determine the technical adequacy of MELCOR,[4] SCDAP/RELAP5 (S/R5),[5] and other severe-accident simulation codes. In this process, the code sponsor specifies both design objectives and targeted applications for the code. The sponsor thus provides a yardstick against which the peer review committee can measure overall technical adequacy of the code.

For the determination of overall adequacy, a collection of plausible phenomena associated with severe-accident behaviors in either boiling-water reactors (BWRs) or pressurized-water reactors (PWRs) must be identified and then ranked for their relative importance in what has been called a phenomena identification and ranking table (PIRT).[6]

This article details a method used to generate a PIRT by examining the relative importance of phenomena and the models of these phenomena used to evaluate a severe reactor accident. The hierarchy-by-interval (HBI) approach can then be used to evaluate severe-accident simulation codes. The HBI approach consists of the following:

1. Identifying and listing the possible phenomena.
2. Checking the existence and adequacy of computer code models for severe-accident phenomena (in our case, we further broke down the phenomena into different time intervals within the accident sequence).
3. Determining conceptually measurable figures of merit related to the design objectives and targeted applications specified by the code sponsor.
4. Having a group of experts rank the phenomena against the figures of merit for each time interval.
5. Generating a table that identifies how well the code calculates phenomena that the experts determined were important.
6. Generating a table that identifies phenomena for which experimental data are needed to better understand the physics involved.

The tables generated by steps 5 and 6 can be termed a PIRT.

An alternative approach that has been used in other industries to generate a PIRT after the first four steps have been completed is the analytic hierarchy process (AHP).[7] This process was used in a direct containment heating application[8] and gave results similar to those based only on expert ranking. The PIRT derived with this process allows the importance of the phenomena to be ranked relative to a figure of merit. It does not provide insight on which phenomena the code models well or indicate phenomena for which a better knowledge of physics is needed as does the HBI method.

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IDENTIFICATION OF DOMINANT SEVERE-ACCIDENT PHYSICAL PHENOMENA

As part of a code peer review process, a list of dominant physical phenomena must first be developed against which the existence, adequacy, and when possible, fidelity of each code model can be assessed. On a generic basis, the various top-level physical phenomena contributing to each phase of severe-accident progression are delineated for both BWRs and PWRs. The importance of an individual phenomenon varies, depending on the specific accident sequence under consideration and the intended application; for example, a code that is intended to simulate in-vessel severe-accident behavior should be applicable to a wide spectrum of severe-accident conditions, including:

1. High- and low- [with respect to the reactor coolant system (RCS)] pressure sequences.
2. Scenarios leading to early [emergency core-cooling system (ECCS) fails early] and late (ECCS fails late) initiation of core degradation.
3. Recoverable accidents.

Typically, severe-accident analyses are performed to better understand the behavior of plant and containment systems during postulated accident conditions. These studies are often conducted in support of probabilistic risk assessments (PRAs) or to provide additional information for regulatory decision making (i.e., resolution of specific safety issues or evaluation of potential severe-accident management strategies). As part of these studies, computer codes are used to evaluate key accident signatures, including some of the following (limited to in-vessel phase only):

1. Timing of key events (core uncovery, lower plenum dryout, vessel breach, containment failure, etc.).
2. Important fission-product attributes (release from fuel, retention within RCS, retention in pools, etc.).
3. Temperatures of RCS structures (lower head, hot leg, steam-generator tubes, etc.).
4. RCS pressure before vessel breach.
5. Mode and location of RCS failure (bottom head, hot leg, steam generator tubes, etc.).
6. Quantity and rate of in-vessel hydrogen generation.
7. Core-debris quantity, composition, temperature, and rate of ejection into containment.

The decomposition proposed here is based on the premise that a complete mechanistic analysis must portray important phenomenological processes that affect any of the key accident signatures during the in-vessel phase of accidents for the following distinct time intervals:

Interval 1: Initial transient, coolant depletion, and heatup interval (before core damage; $T_{vessel\ exit} \leq T_{saturation}$).
Interval 2: Core uncovery interval (intact geometry; $T_{core} < 1500$ K).
Interval 3: Melt relocation and slump interval (substantial damage; $T_{core} > 1500$ K).
Interval 4: Core-debris material inside the lower plenum interval (late in-vessel phase).

For each interval, key phenomenological issues affecting the evolution of the accident sequence are delineated. For the process to remain tractable, detailed subissues resulting from higher order phenomena associated with the interaction of various physical and chemical processes are intentionally not shown. This does not mean that the dominance of some of these phenomena should be ignored during code review.

Figures 1 to 4 show the hierarchical decomposition of the interval-dependent phenomena dominant in a hypothetical severe accident in a typical PWR plant. To generate these figures we started with the MELCOR peer review[1] lists and added potential phenomena that were of interest. A corresponding flowchart for a BWR plant is given in Appendix C of Ref. 2. Figure 1 includes two box diagrams. The upper diagram depicts the three dominant phenomena occurring during the initial transient: (1) the fission and decay heat source to the core, (2) the ability of the structures to remove heat, and (3) the reactor coolant system thermal–hydraulics.

The lower diagram details the five dominant phenomenological areas affected during the core uncovery interval: (1) the phenomena necessary to describe the core state—how the fuel rods have heated and deformed, (2) the fission and decay heat source to the core, (3) the initial release of fission products and how they interact with the RCS and any water pools, (4) the basic thermal–hydraulics and two-phase flow associated with core uncovery, and (5) the oxidation of the metals in the core below 1500 K. In both the upper and lower diagrams, the hydrodynamics and (fluid) transport box has a double line surrounding it to indicate that it will be detailed in Fig. 3.

Figure 2 shows the hierarchical decomposition diagram for interval 3, the time from when the core has reached 1500 K to the time when significant debris is within the lower plenum. This is a wide-ranging time frame that depends heavily upon the accident scenario

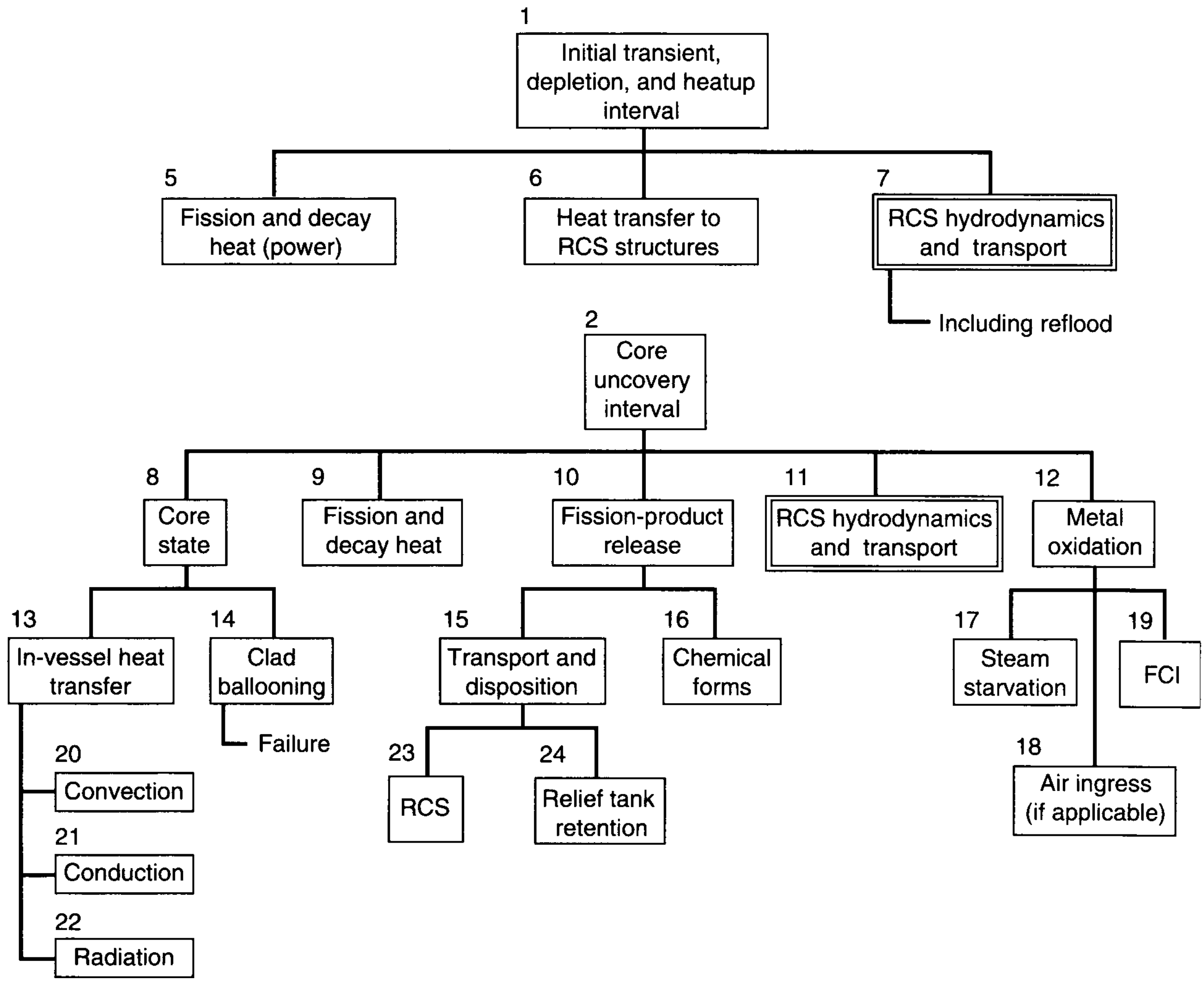

Fig. 1 Dominant phenomena for initial transient and core-uncovery intervals in a pressurized-water reactor plant. Note: FCI is fuel–coolant interaction and RCS is reactor coolant system. [The numbers correspond to the severe accident phenomena of interest. The single boxes represent major phenomena and the double boxes represent major phenomena that will be detailed in Fig. 3. This convention applies to Figs. 1 to 4.]

but is intended to describe important phenomena occurring within the core zone. It has been divided into nine phenomenological areas, of which RCS hydrodynamics and (fluid) transport and melting and freezing are detailed in Fig. 3. The other seven areas are (1) the fission and decay heat source to the core; (2) the transfer of heat from fuel rods to control blades and structures; (3) the oxidation of metals in this high-temperature environment; (4) the formation of eutectics, which causes a lower melting temperature mixture that can relocate and potentially form blockages; (5) the behavior of the fission products as they are released from within the fuel and transported through the RCS to the containment; (6) the control blade failure, relocation, and interaction with other core materials; and (7) the relocation and slumping of in-core material as blockages fail.

Figure 3 shows the hierarchical decomposition diagrams for the RCS hydrodynamics and (fluid) transport from all intervals as well as the in-core melting and freezing phenomena from interval 3. The RCS hydrodynamics and transport diagram is divided into four areas: (1) the phenomena associated with reflooding the core with water, (2) the calculation of whether natural circulation is full loop or just within the core and upper plenum, (3) the discharge and blowdown calculation,

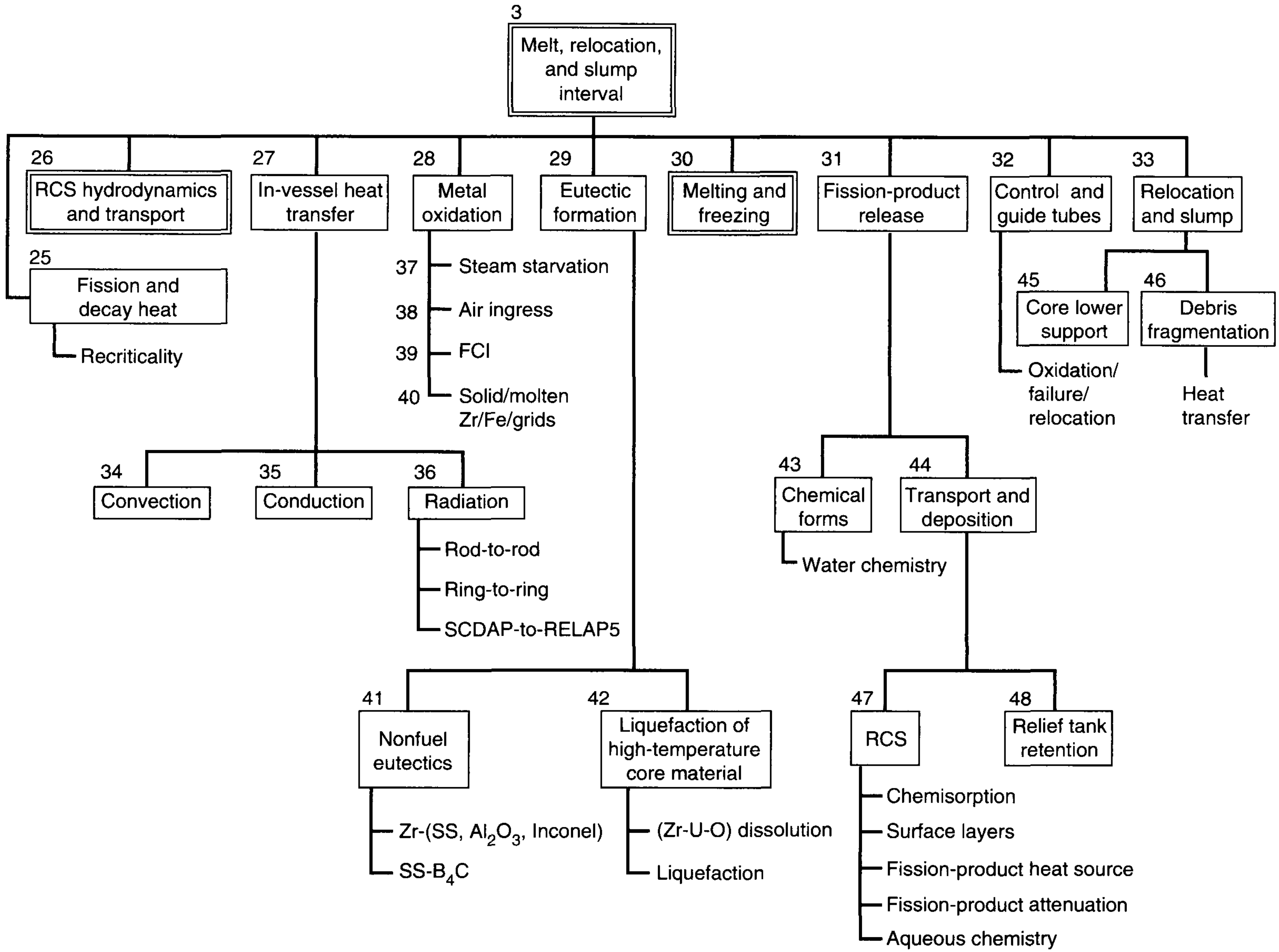

Fig. 2 Dominant phenomena for melt relocation and slump interval in a pressurized-water reactor plant. Note: SS is stainless steel, FCI is fuel–coolant interaction, and RCS is reactor coolant system.

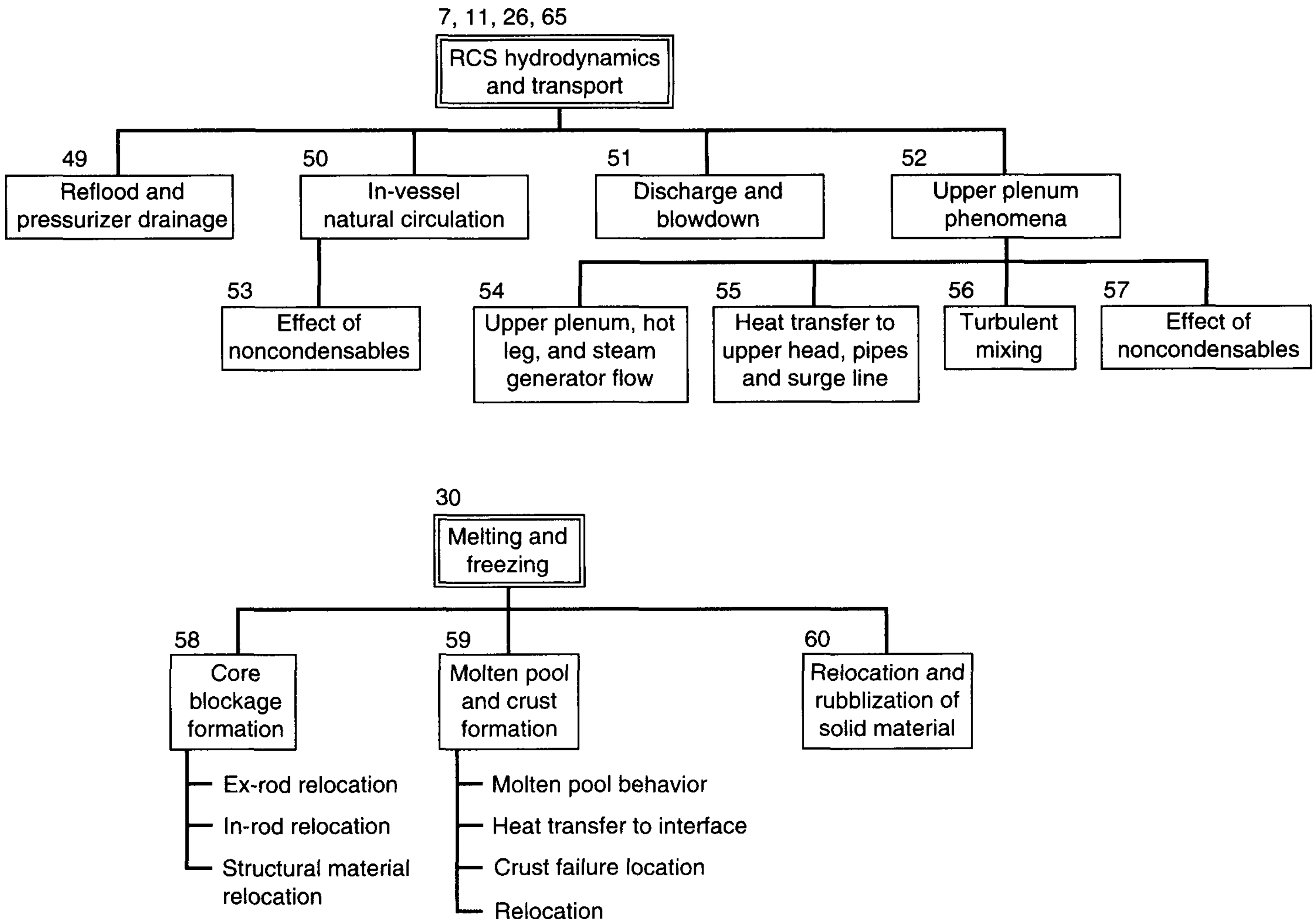

Fig. 3 Dominant phenomena in reactor coolant system hydrodynamics and melting and freezing decompositions. Note: RCS is reactor coolant system.

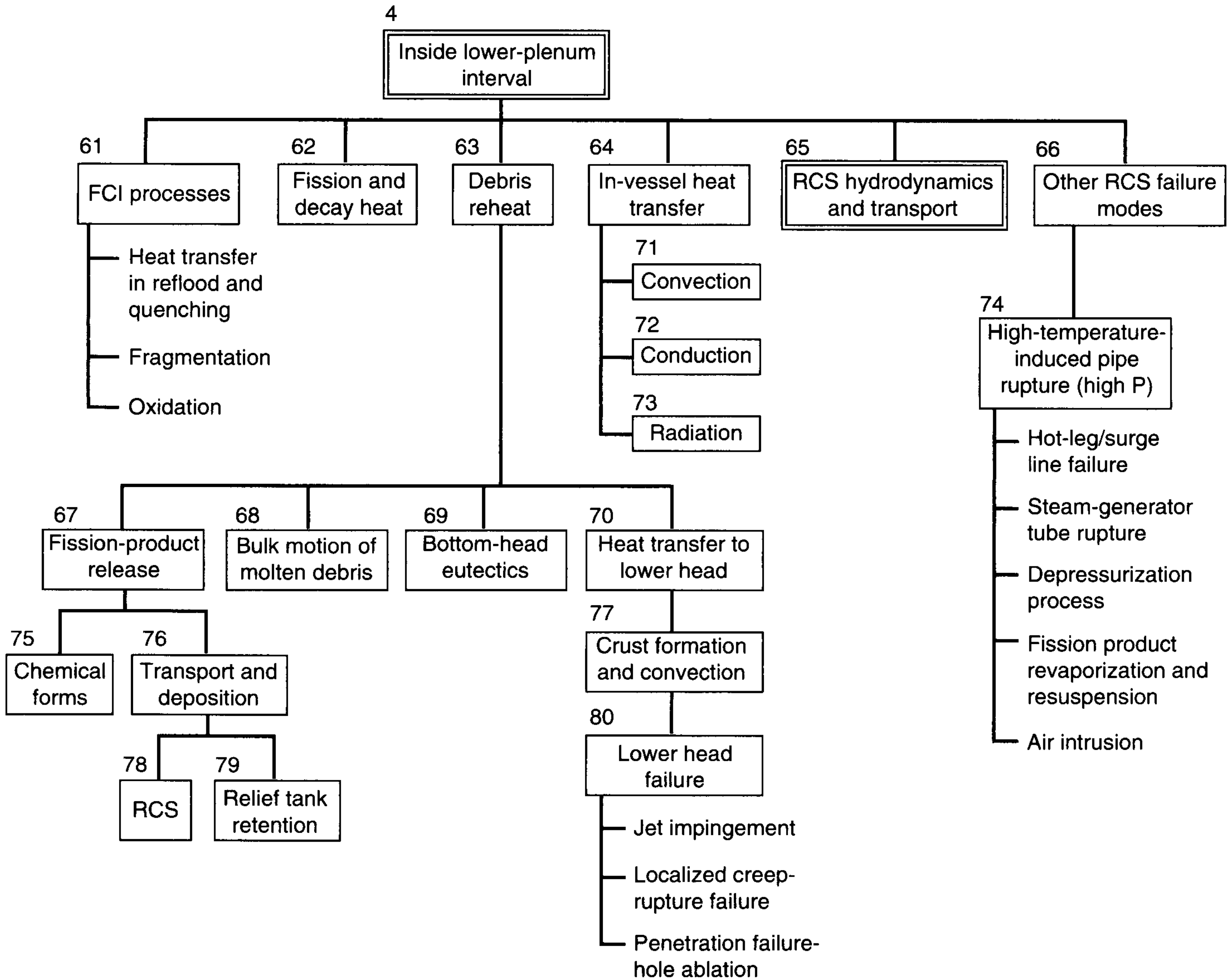

Fig. 4 Dominant phenomena for core debris inside the lower plenum interval in a pressurized-water reactor plant. Note: FCI is fuel–coolant interaction, RCS is reactor coolant system, and P is pressure.

and (4) all upper plenum phenomena. The lower box diagram separates the in-core melting and freezing area into three parts: (1) the basic relocation of fuel material within the rods and external to the rods as well as the interaction with other core materials and the formation of a blockage; (2) how material piled up behind the blockage forms a pool and how the pool interacts with its surroundings; and (3) how fuel rods can be turned into rubble (for example, by reflooding) and relocated downward.

Figure 4 shows the hierarchical decomposition diagram for the final interval in which the lower plenum and vessel lower head failure phenomena are considered. This interval has been divided into six phenomenological areas; the RCS hydrodynamics and (fluid) transport area was detailed in Fig. 3. The remaining five areas are (1) the interaction of the relocating core material with either the water in the lower plenum or reflood water; (2) the fission and decay heating of the relocated core debris; (3) how the debris heats, releases fission products, and interacts with its surroundings; (4) the heat transfer between the debris, the rest of the in-core structures, and the vessel; and (5) the effect of other RCS failures and/or depressurization processes.

RANKING SEVERE-ACCIDENT PHENOMENA USING THE HBI APPROACH

A systematic and detailed ranking of the severe-accident phenomena can be undertaken to identify both the physical processes important to the outcome of a severe accident and the code models that exist to represent those physical processes. The HBI approach combines the results of the review of the technical adequacy of the phenomenological models in the code with the decomposition of the phenomenological block diagrams presented in the previous section. With this decomposition scheme, some phenomena may occur within several intervals with different levels of importance and technical adequacy.

The initial task is to identify a level of importance for each of the severe-accident phenomena detailed in Figs. 1 to 4. We must also determine not only which phenomenological areas or parts of an area are more important in determining the effects of a severe accident but also why they are important. To do this, several figures of merit are used against which the phenomena can be rated (high, medium, or low) by knowledgeable reviewers. The figures of merit chosen should be conceptually measurable and have a significant effect upon containment phenomena occurring during a severe accident. The following figures of merit used in the S/R5 review are taken from the severe-accident signatures detailed earlier.

- Source term: The timing, magnitude, and condensed vapor phase of fission-product release to the containment.
- Hydrogen generation: The timing and release rate of hydrogen to the containment.
- Melt ejection characteristics: The composition of the corium released from the vessel. This includes the mass fractions, melt fractions, and the temperature of the ejected material.
- RCS failure: The timing and location of failure of the RCS. This is primarily a function of the RCS temperature distribution and pressure history.

The following additional figure of merit was added for the experimental analysis area.

- Peak temperature: This relates the experimental data to the code results and implies a temperature history.

In the S/R5 review, each of the phenomena from Figs. 1 to 4 was listed, and each member of the committee gave a rating against each figure of merit. These ratings were then averaged and a final table generated. Table 1 gives an example for the dominant phenomenon from the lower plenum interval from Fig. 4. As indicated, the phenomenon and its identifying number are in column 1, followed by a column for each of the averaged ratings (high, medium high, medium, medium low, and low) for the five figures of merit, and a final column showing the knowledge of physics as determined by the committee. The knowledge-of-physics column represented what the review committee thought of the state of the art for a particular phenomenon (i.e., whether the physics were understood, questionable, or poorly understood). Because the committee came from diverse backgrounds, their views of a particular phenomenon could be different. To help come to an acceptable viewpoint, each member voted, and a discussion was held on each phenomenon in each time interval.

Six phenomenological areas are numbered 61 to 66, listed in Fig. 4. Some of the areas are further subdivided; others are not, but all labels are listed separately in Table 1. Therefore, in Table 1 there are four entries with the number 61 to indicate the subdivisions. Note that the first entry, fuel–coolant interaction (FCI) processes—debris fragmentation, was determined to be

Table 1 (Interval 4) Figure-of-Merit Ranking for Lower Plenum[a]

	Phenomena	Source term	Hydrogen generation	Melt ejection characteristics	RCS failure	Peak temperature	Knowledge of physics
61	FCI processes—debris fragmentation	H	H	MH	M	M	P
61	FCI melt/debris heat transfer to water	L	H	MH	MH	H	P
61	FCI processes—metal oxidation	ML	H	MH	ML	H	Q
61	FCI processes—fission-product release	H	L	L	L	L	Q
62	Fission heat	L	ML	ML	L	ML	U
62	Decay heat	MH	MH	MH	M	H	U
63	Debris heatup process lower plenum molten pool formation	L	L	H	H	H	P
65	RCS hydrodynamics and transport	H	H	H	H	H	U
66	RCS failure	MH	ML	MH	H	ML	U
67	Fission-product release	H	L	L	L	L	P
68	Bulk motion of molten debris	L	L	ML	MH	ML	Q
69	Bottom head eutectics	L	L	M	H	M	Q
70	Debris/molten pool heat transfer to head	L	L	M	H	M	U
71	Convective heat transfer	M	M	MH	H	H	U
72	Conduction	M	M	M	MH	MH	U
73	Radiation	L	M	MH	M	H	U

[a]FCI, fuel–coolant interaction; RCS, reactor coolant system. H, MH, M, ML, and L refer to high, medium high, medium, medium low, and low, respectively. P, Q, and U refer to poorly understood, questionable, and understood, respectively.

highly important from the source term and hydrogen generation points of view, moderately important in determining the RCS failure and peak temperature, and between highly important and moderately important in affecting the melt ejection characteristics. The fragmentation process was also considered to be poorly understood. Other parts of this same phenomenological area (i.e., FCI processes—fission-product release) could have a better knowledge base but a ranking of lesser importance.

Not all the figures of merit can be rated equally, so the committee grouped an average of the hydrogen generation, melt ejection characteristics, and RCS failure columns and called the grouping a core damage importance criterion. The committee also eliminated the peak temperature column from final consideration and left the source term criteria separate. Thus, in terms of importance criteria, two values are listed in the final consolidated table: the source term for radiological consequences and core damage for all other consequences.

The last task was to generate a consolidated table and then sort and rank the importance of a phenomenon relative to the figures of merit. This required the code developers to identify which code models simulate these phenomena. Tables 2 to 5 show part of the final consolidated table from the S/R5 review. As in the previous table, the phenomena are listed in column 1; the location of the write-up of the S/R5 model that corresponds to the phenomena is listed in column 2; the knowledge of the physics obtained from the committee discussion of each model is listed in column 3; the level of physics incorporated into the model is given in column 4; the level of importance relative to source term or core damage is listed in column 6; and the current validation status is given in column 7. Column 5 details the technical adequacy of the S/R5 model for the listed phenomena. A detailed discussion of this topic is contained in the peer review document, but generally speaking, a value of 1 to 3 is acceptable. A value of 6 or 7 indicates that there was no model in the code (but the importance of the model to determine core damage is low). A value of 4 or 5 indicates that modeling work is needed. Each model was reviewed and given a technical adequacy rating by one of the committee, and that evaluation was then reviewed by another member of the committee. During the presentations of the reviews, a consensus was developed on how adequately the S/R5 models the phenomena.

As shown in Tables 2 and 3, none of the phenomena for intervals 1 or 2 was deemed to be of high importance. This is to be expected because, when core damage is the critical figure of merit, the initial phases of the accident (the period when the temperatures remain below 1500 K) would be less important than the later phases. Also note in Table 2 the technical adequacy value of 7 given to both the fission and decay heat

Table 2 Hierarchy by Interval (Interval 1) Ranking of Phenomena for Initial Transient Heatup $T_{vessel\ exit} \leq T_{saturation}$

	Phenomena	S/R5[a] code model		Knowledge of physics	S/R5 physics	Technical adequacy	Importance[b] ST	CD	Validation status
5	Fission heat	2.2	Nuclear heat	Understood	0th order	7	M	ML	Validated
		2.5	Fuel-state models RELAP5	Understood	0th order	7			Validation possible
5	Decay heat	2.22	Fission-product heat	Understood	0th order	7	ML	L	Validated
		2.5	Fuel-state models	Understood	0th order	7			Validation possible
6	Heat transfer to RCS structures	RELAP5				Not reviewed	L	L	
7	RCS hydrogen and transport	RELAP5				Not reviewed	L	L	

[a]SCDAP/RELAP5.
[b]ST and CD refer to source term and core damage, respectively. M, ML, and L refer to medium, medium low, and low, respectively.

Table 3 (Interval 2) Ranking of Phenomena for Core Uncovery $T_{core} < 1500$ K

	Phenomena	S/R5[a] code model		Knowledge of physics	S/R5 physics	Technical adequacy	Importance[b] ST	CD	Validation status
11	Hydrodynamics and transport	RELAP5				Not reviewed	M	MH	
12	Metal oxidation Zircaloy rods (see also 12 to 19)	2.1	Material oxidation	Understood	1st order	1	ML	MH	Validated
9	Decay heat	2.22	Fission product	Understood	0th order	4	MH	M	Validated
		2.5	Fuel-state models	Understood	0th order	4			
20	Convection	RELAP5				Not reviewed	M	M	
11	Hydrodynamics and transport steam-generator tube rupture	RELAP5				Not reviewed	M	M	

[a]SCDAP/RELAP5.
[b]ST and CD refer to source term and core damage, respectively. MH, M, and ML refer to medium high, medium, and medium low, respectively.

models. This is an example of where the model implemented did not calculate the phenomena correctly as detailed in Ref. 2. The use of the default table look-up, however, was considered adequate for severe-accident analysis. Table 4 relates to interval 3, where the core undergoes melt relocation and slumping. In this phase, 12 phenomena were determined to be important. In interval 4, the lower head phenomenon, whose importance was detailed in Table 1, has 12 important phenomena, as shown in Table 5. Note that the RELAP5 models (depressurization, RCS hydrodynamics, and convection) maintain a critical importance throughout the calculation of a severe accident. As shown in the table, it is necessary to validate RELAP5 models used during the later phases of an accident analysis.

Tables 2 to 5 can be used to provide a rationale for further work in the severe-accident area. Those phe-

Table 4 (Interval 3) Ranking of Phenomena for Melt Relocation and Slump $T_{core} > 1500$ K

	Phenomena	S/R5[a] code model		Knowledge of physics	S/R5 physics	Technical adequacy	Importance[b] ST	Importance[b] CD	Validation status
49	Reflood	2.25	Severe-accident thermal–hydraulics	Questionable		4	H	H	Insufficient data
34	Convection	RELAP5				Not reviewed	H	H	
30	Melting and freezing (see also 58 to 60)	2.4	Effective materials properties				M	H	
58	Core blockage formation	2.9	Liquefaction, flow, and solidification	Questionable	0th order	4	M	H	Inadequate implementation
		2.13	Core-region debris modeling	Poor	0th order	3			Insufficient data
58	Core blockage formation ex-rod relocation	2.9	Liquefaction, flow, and solidification	Poor	0th order	3	M	H	Insufficient data
		2.13	Core-region debris modeling	Poor	0th order	3			Insufficient data
58	Core blockage formation in-rod relocation	2.26	Additional models being developed or upgraded	Poor		6	M	H	Insufficient data
28	Metal oxidation Zircaloy rods (see also 37 to 40)	2.1	Material oxidation	Understood	1st order	1	M	H	Validated
45	Relocation and lower core plate	2.14	Core slumping	Poor	0th order	3	M	H	Insufficient data
		2.15	Lower plenum debris heatup	Poor	1st order	3			
		2.26	Additional model being developed or upgraded						
33	Crucible relocation and slump (see 45 to 46)						ML	H	
50	In-vessel natural circulation (see also 53)	RELAP5				Not reviewed	H	MH	
32	Channel box and control rods relocation	2.11	Control rod and core structure	Questionable	0th order	4	H	MH	
46	Relocation and debris fragmentation	2.13	Core-region debris modeling	Poor	0th order	3	M	MH	Insufficient data
42	Eutectic Zr–U–O dissolution $T < T_{melt}$	2.9	Liquefaction, flow, and solidification	Questionable	0th order	4	ML	MH	Validation possible
		2.21	Materials properties	Understood	1st order	1			

[a]SCDAP/RELAP5.
[b]ST and CD refer to source term and core damage, respectively. H, MH, M, and ML refer to high, medium high, medium, and medium low, respectively.

Table 5 (Interval 4) Ranking of Phenomena for Lower Plenum

	Phenomena	S/R5[a] code model	Knowledge of physics	S/R5 physics	Technical adequacy	Importance[b] ST	Importance[b] CD	Validation status
65	RCS hydrodynamics and transport (see 26)	RELAP5	Not reviewed			H	H	
74	Depressurization	RELAP5	Not reviewed			H	MH	
61	FCI processes—debris fragmentation	Input 2.14 Core-slumping model	Poor	Features	3	H	MH	Insufficient data
66	RCS failure	Separate calculation	Understood	0th order	4	MH	MH	Validated
62	Decay heat	2.15 Lower-plenum debris heatup	Understood	No features	5	M	MH	Validation possible
71	Convection	RELAP5	Not reviewed			M	MH	
61	FCI processes—metal oxidation		Questionable		5	ML	MH	Insufficient data
61	FCI processes—debris heat transfer to water	2.14 Core-slumping model	Poor	No features	6	L	MH	Insufficient data
63	Debris reheat lower-plenum molten pool formation	2.15 Lower-plenum debris heatup	Poor	0th order	3	L	MH	Insufficient data
69	Bottom-head eutectics		Questionable		5	L	MH	Validation possible
77	Lower-plenum crust behavior	2.15 Lower-plenum debris heatup	Questionable	0th order	4	L	MH	Insufficient data
70	Heat transfer to lower head	2.15 Lower-plenum debris heatup	Understood	1st order	1	L	MH	

[a]SCDAP/RELAP5.

[b]ST and CD refer to source term and core damage, respectively. H, MH, M, ML, and L refer to high, medium high, medium, medium low, and low, respectively.

nomena judged to be of high importance or of moderately high importance should be examined for completeness [for example, in interval 3 (Table 4) the relocation phenomena were generally considered to be poorly understood but were of high importance]. A rating of poorly understood in the knowledge of the physics implies a need for fundamental experiments and for scoping analysis models that capture the essence of the phenomena. Thus work that has the goal of improving the understanding of the physics should be emphasized. A low rating in technical adequacy (4 or 5) implies a need for better models, particularly when the physics is known, and a low rating for validation status implies a need for further assessment work. Only the phenomena of higher importance are shown in Tables 2 to 5. A more detailed listing of all the dominant phenomena is given in Appendix D of Ref. 2 for the SCDAP/RELAP5 review.

DISCUSSION

The technical adequacy of a severe accident phenomenological model strongly depends on the interval of the severe accident to which it is applied; for example, in the SCDAP/RELAP5 peer review, we found that many code models were technically adequate during the early intervals of an accident but were deemed technically inadequate as an accident progressed into the later intervals where core degradation, relocation, and possible vessel failure might occur.

On the basis of a resorting and ranking of the dominant phenomena and associated code models described in this paper, it was possible to prioritize the inadequate models needing improvement where the knowledge of the physics was deemed adequate. For the SR/5 code, the most important model needs were the following:

1. Interface between SCDAP and RELAP5. The

code used inappropriate heat transfer and friction correlations for both reflood and convective flow over degraded geometries (phenomenon 34 in Table 4 and phenomenon 71 in Table 5).

2. Decay heating in debris beds and molten pools. This was a case in which there was no model for a major heat source (phenomenon 62 in Table 5).

3. Zirconium dissolution of UO_2 resulting from eutectic formation. The models used did not employ state-of-the-art formulations for this phenomenon (phenomenon 42 in Table 4).

Models that could be improved where the knowledge of the physics was more questionable include the following:

1. Reflood, specifically the interaction of water with high-temperature structures (phenomenon 49 in Table 4).
2. Relocation of control material and core structures (phenomena 32 and 58 in Table 4).
3. Core blockage formation and oxide shell failure (phenomenon 58 in Table 4).

Areas in which the knowledge of the physics should be improved include the following:

1. Relocation and blockage formation by fuel rods (phenomenon 58 in Table 4).
2. In-core and lower plenum molten pool formation (phenomena 45 and 46 in Table 4 and phenomenon 63 in Table 5).
3. Lower plenum FCI processes, including fragmentation and interaction with water in the lower plenum (phenomenon 61 in Table 5).

The reader should be aware that the suggested model improvements are based on a code review performed in 1992, and many code deficiencies have subsequently been addressed by the code developers.

CONCLUSIONS

Resource and technical constraints do not allow a complete resolution of all the severe-accident uncertainty and phenomenological issues. Therefore resolution of issues within the regulatory and licensing framework requires a focused assessment of significant phenomenological questions with respect to their impact on public health and safety.

Hierarchical decomposition of physical and chemical phenomena related to accident progression provides a structured approach for establishing the necessary link between the state of the art and its interpretation within a computer code and between individual phenomena and severe-accident sequences. It thereby relates the knowledge and importance of particular phenomena to the goals and purpose for which the code was intended.

Establishment of the relative ranking of dominant phenomena as exemplified in this paper can help focus theoretical (i.e., simulation) models and experimental research needs to better achieve rational objectives.

The hierarchy-by-interval approach developed in this paper is a process to determine the technical adequacy of the SCDAP/RELAP5 code models on the basis of a systematic ranking of dominant severe-accident phenomena. This approach helps focus the peer review process on those modeling and phenomenological issues which have the greatest potential impact (at least qualitatively) on severe-accident consequences (i.e., core damage and radiological source terms). It has demonstrated its effectiveness by identifying important phenomena within a particular code that need an improved formulation.

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RELAP5/MOD3 Code Coupling Model

By R. P. Martin[a]

Abstract: *A new model has been built into RELAP5/MOD3 to facilitate coupling RELAP5/MOD3 and other computer codes. The new model has been designed to support analysis of the new advanced reactor concepts. Its user features rely solely on new RELAP5 "styled" input and the Parallel Virtual Machine software, which facilitates process management and distributed communication of multiprocess problems. RELAP5/MOD3 manages the input processing, communication instruction, process synchronization, and its own send-and-receive data processing. The flexible model requires that an explicit couple be established, rather than a more accurate implicit model, to update boundary conditions at discrete time intervals. Two test cases are presented that demonstrate the functionality, applicability, and issues involving the use of this model.*

The primary mission that supported the development of this model was the coupling of RELAP5/MOD3 (Ref. 1), a best-estimate thermal–hydraulic systems code, and CONTAIN,[2] a containment analysis tool. The motivation for the union of these two computer codes stems from the unique safety analysis challenge presented by the new Advanced Light-Water Reactor (ALWR) conceptual designs. Incorporated into many of these designs are requirements for long-term passive cooling systems, which integrate both mechanisms in the main reactor coolant system and in the containment. Westinghouse's AP600 and General Electric's Simplified Boiling-Water Reactor (SBWR) are examples of two designs that meet this description.

Proof of principle that RELAP5/MOD3 could be coupled with CONTAIN was performed at Pennsylvania State University.[3] This work demonstrated that the state-of-the-art best-estimate codes could be linked and could generate very meaningful results. The RELAP5/MOD3 code coupling model evolved from this project to feature a generic infrastructure within RELAP5/MOD3 for defining links between RELAP5/MOD3 and another computer code. The implementation of this concept, as described in the following sections, extends the previous work by addressing the lessons learned from the original effort and by adding robustness to the coupling.

The code-coupling model exploits the Parallel Virtual Machine (PVM) software[4] developed at the Oak Ridge National Laboratory for the Department of Energy. The PVM software was designed to provide multiprocessing capabilities on a loosely coupled network of diverse computer systems. The primary roles of PVM, as applied to the RELAP5/MOD3 code-coupling model, are in the process management, interprocess communication, and synchronization capabilities it offers. These routines manage the identification of parallel processes, the timing of data delivery from one code to the other, and the transmission of data from one code to the other.

The code-coupling link described with this model can be used to define an "explicit couple" with RELAP5/MOD3. An explicit couple implies that the calculation solutions of RELAP5/MOD3 and the coupled code are performed independently with respect to the system model described in the input model. Data from a code are introduced into the other code through static or dynamic boundary conditions imposed on the system models. Application of an explicit coupling model—although not as accurate as an implicit method that simultaneously solves the solution matrices of the complete problem described by separate system models—allows for the general application of a coupling model. In the limit where data are exchanged every time step, the coupling can be considered as semi-implicit; however, this situation is computationally intensive and not likely to be a practical use of this model.

Integration of the strengths of other sophisticated analysis tools and new phenomenological models will complement the sophisticated models inherent in RELAP5/MOD3. This added capability will allow for improved analysis and simulation of thermal–hydraulic systems by providing a means for applying phenomenological models of systems that are beyond the scope of RELAP5/MOD3. New models also can be tested through this link quickly while maintaining the integrity

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of the RELAP5/MOD3 coding. An additional benefit of this feature is that it allows for the exploitation of dual processor machines that will enhance performance of a calculation.

SOFTWARE DESIGN

The software design philosophy for implementing a code-coupling model into RELAP5/MOD3 was to develop the model considering how the human input model developer and analyst would interface with it. Considerable effort went into the design of the user interface to ensure ease of use. The result is that the input model developer and code analyst must only learn how to provide coupling input to the RELAP5/MOD3 input models and how PVM works with processes.

All software development for this project was performed solely on the Cray YMP supercomputer at the Idaho National Engineering Laboratory because the CONTAIN code has not been fully tested and approved by the U.S. Nuclear Regulatory Commission for use on a workstation platform. The software design does not include platform-specific routines that would prevent the use of the new model on a workstation. Tests have been conducted on a Sun workstation running under SunOS.

The Role of PVM Software

The PVM software was designed to provide multiprocessing capabilities on a loosely coupled network of diverse computer systems. The application of the PVM software to the RELAP5/MOD3 code-coupling model provides process management, interprocess communication, and synchronization capabilities. These routines will manage the identification of parallel processes, the timing of data delivery from one code to the other, and the transmission of data from one code to the other.

The process management routines in PVM that are used in the coupling model include functions that identify processes for parallel execution and spawn individual processes. The identification of processes for parallel execution involves establishing a link that can be referenced for all communication between parallel processes. The spawning of a process begins execution of another process and begins any further communication between processes.

The data transfer routines in PVM provide the message-passing feature necessary for communicating RELAP5/MOD3 data and data from another code. Within PVM, a message destination is referenced with data transmission routines during execution. Query routines are also available to monitor how the communication is proceeding. PVM version 3.1 was used with the RELAP5/MOD3 code-coupling model.

Coordination Strategy

The design of a code-coupling model in RELAP5/MOD3 demands that some overhead be performed for this to be a useful feature. Performance of such overhead distinguishes RELAP5/MOD3 as the "parent" process in any coupled calculation. The actual "parent" responsibilities of RELAP5/MOD3 are minimal. They involve reading information provided by input, executing the "child" process, determining information required by the "child," sending that information, and then releasing the link following the calculation. A data stream containing information on the frequency of communication and the structure of the data transmission data streams before initiation of a calculation is sent to the child process. Following this step, both processes run independently, pausing for data transmission at input-prescribed times. Both the parent and child processes are responsible for the collection of data to be transferred and the integration of received data into respective solution schemes. Synchronization is managed by requiring confirmation of receipt following data transmission.

Data Compilation, Manipulation, and Integration in RELAP5/MOD3

Because RELAP5/MOD3 is the parent process when coupled with another code, it is responsible for determining the data shared between RELAP5/MOD3 and the other code and conveying that information to the other process. This information must contain a RELAP5/MOD3 source type (i.e., the RELAP5/MOD3 variable or system state, such as pressure and temperature), volume number (the location in the analyst's model), labels that describe the equivalent information in the child process, and a message tag that specifically identifies the information being sent. All this information comes from the RELAP5/MOD3 input file.

The RELAP5/MOD3 source type/volume number pair defines the source or sink of data going to and from RELAP5/MOD3, respectively. The information is formatted like "minor edits" used regularly in a typical RELAP5/MOD3 model. The advantages of using RELAP5/MOD3 variables directly are (1) the reduction of specific hardwired coding into RELAP5/MOD3, (2) the flexibility in being able to define control variables that are not normally available, and (3) the general extension of this coupling model for use with any code. The data are sent to the other code sorted by message tag. Data received from the child process are incorporated into a time-dependent volume or time-dependent junction, depending on the kind of information received.

The child process simply must act on the data it receives from RELAP5/MOD3. RELAP5/MOD3 is responsible for sending the data needed by the child process to use in the child process's calculation. As determined by the RELAP5/MOD3 input file, a data stream is sent from RELAP5/MOD3 to the child process. The child process receives this information and incorporates the data appropriately for the problem being solved, as defined by the labels given in the input. Conversely, the child process must gather the data RELAP5/MOD3 needs and send it to RELAP5/MOD3.

Input Format

The input format contains information on which process to start, the frequency of data transmission for both sending and receiving data for the parent, the parent-to-child link descriptions, and the child-to-parent link descriptions. The child process name identifies the child process. Data transmission frequency can be provided as a function of time through the inclusion of additional input cards, which gives the user flexibility to perform coupled calculations more efficiently by eliminating unnecessary communication between processes. Separate lists describe (1) exactly what is sent and how to send it to the child process and (2) what information is received from the child process and where to put it. The 20900000 card number series has been created for this new feature. An example follows in Fig. 1.

The 20900000 card, or card 0, is reserved for the name of the executable to be coupled with RELAP5/MOD3. In this example, the executable is CONTAIN. Cards 01–99 are used for expressing the frequency of communication between the two codes. In the previous sample input, RELAP5/MOD3 and CONTAIN exchange data every 0.1 s until the calculation advances to the 10-s mark; data are then exchanged every 1.0 s until the 50-s mark. Cards 1001–1999 are used to identify system states calculated by RELAP5/MOD3 and identify where these data should be used in the child process. The last value is the message tag.

DESCRIPTION OF NEW CODING

The existing coding in RELAP5/MOD3 conforms to the FORTRAN 77 standard, and all modifications and extensions to the existing coding adhere to the FORTRAN 77 standard and the existing style and idiom of RELAP5/MOD3. Additionally, a RELAP5/MOD3 executable must include the library of routines that make up the PVM software. Figure 2 is a flowchart depicting how RELAP5/MOD3 and a generic child process are coupled.

Code modifications to RELAP5/MOD3 and a child process are isolated to single calls to new subroutines

```
20900000   contain.x
20900001   0.1    0.0    10.0
20900002   1.0    0.0    50.0
* RELAP5 Outputs to CONTAIN
20901001      mflowj        1000000      mflow        1      atm    1      101
20901002      ufj           1000000      enthalf      1      atm    1      101
20901003      mflowj        1000000      mflow        1      atm    1      401
20901004      ugj           1000000      enthalf      1      atm    1      401
* RELAP5 Inputs from CONTAIN
20902001      p             1020000      pgas         1      atm    1      101
20902002      tempg         1020000      tgas         1      atm    1      101
20902003      uf            1020000      uf           1      atm    1      101
20902004      ug            1020000      ug           1      atm    1      101
```

Fig. 1 Sample code coupling input.

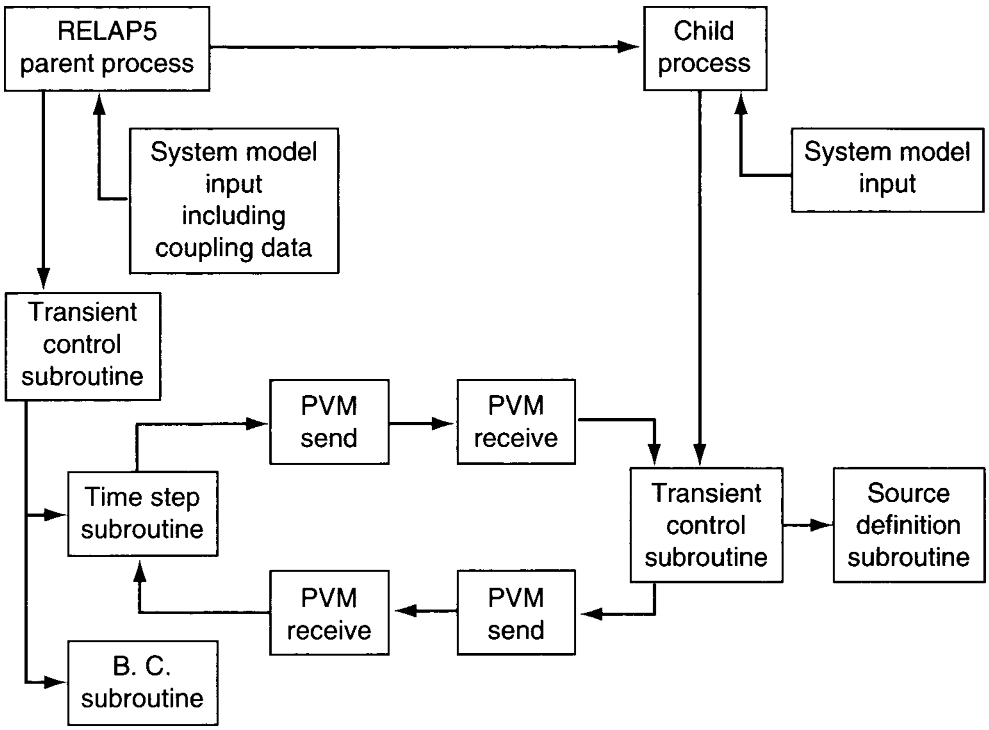

Fig. 2 RELAP5/MOD3 generic code coupling flowchart. [PVM is Parallel Virtual Machine. B. C. is boundary conditions.]

that involve the initialization of the coupling and the reading, interpreting, and following the instruction of the coupling data. RELAP5/MOD3 handles all input processing and the initialization of the coupling calculation, which involves enrolling RELAP5/MOD3 as a process under PVM and spawning the second process under PVM. Most other subroutines used in both RELAP5/MOD3 and the child process are nearly identical in function. The two main subroutines, PVMSND and PVMRCV, are called to monitor the data exchange frequency, create or interpret a data stream on the basis of the coupling information, and exchange the data stream at the specified interval times. At the first-time step of any coupling calculation, PVMSND in RELAP5/MOD3 provides the child process with the start and end times of the calculation, the number of send and receive messages, the frequency of communication information, and the specific messages. During the calculation, PVMSND and PVMRCV determine if, at any given time step, it is time to exchange data between processes. If so, data are exchanged. This procedure requires synchronization; therefore every send call is followed by a receive call verifying that the data were sent properly. Separate subroutines manage the implementation of data received from one process to another. Additionally, unique subroutines have been implemented to provide error checking during sending or receiving between processes. The subroutines provide a "time out" if a process has not responded within a specified time interval, and they also check to ensure that PVM is still activated.

CAPABILITY, LIMITS, AND EXPANDABILITY OF RELAP5-BASED CODE COUPLING

Coupling RELAP5/MOD3 and a child process in this configuration creates a powerful new tool for nuclear power plant systems analysis. This configuration permits a wide range of flexibility for establishing links between two codes with very specific coupling information. The coupling data input tells the two codes (1) exactly what data to transfer, (2) when to transfer the data, and (3) how to use the data when received by the other process; however, this configuration does not facilitate the coupling of the simultaneous equations in both codes to achieve the best accuracy possible. Instead, the data received by a process are integrated as constant boundary conditions (i.e., an explicit couple), which can introduce error that is dependent on the frequency of communication. In the extreme case in which communication between the codes occurs every time step (i.e., a semi-implicit couple), this error may be negligible; however, it may be a computationally intensive situation that would not be attractive from a productivity standpoint. Conversely, the use of very large time steps also would not be attractive from an accuracy standpoint. This situation requires that the analyst perform time-step sensitivity calculations to assess the accuracy benefits of smaller time steps vs. the productivity benefits of larger time steps. Because RELAP5/MOD3 control variables are available to send to a child process, corrections can be applied to data being sent to reduce this error. A possible future feature of this coupling might include a time-step control based on information passed from RELAP5/MOD3 through control variables.

TESTING, VERIFICATION, AND EXPERIENCE WITH MODEL

The new coding has been verified through test cases analyzing the coupled performance of RELAP5/MOD3 and the Sandia National Laboratories CONTAIN code, two RELAP5/MOD3 processes, and RELAP5/MOD3 and a simple accumulator model. Two cases were performed with RELAP5/MOD3 and CONTAIN. The first case was a simple pressure-vessel blowdown into a small containment. A more robust case was performed analyzing a main steam line break (MSLB) in the General Electric SBWR. The other cases were simpler and demonstrated the wide applicability and user issues of the coupling model. Discussion of the two RELAP5/MOD3 and CONTAIN tests should adequately demonstrate the functionality of the RELAP5/MOD3 code coupling model.

RELAP5/MOD3 Coupled with CONTAIN: Pressure-Vessel Blowdown

This test involved the blowdown of a pressure vessel at 2.0 MPa (290 psia) containing saturated liquid water. The RELAP5/MOD3 model is coupled with a single-volume CONTAIN model that is linked to receive the mass and enthalpy from RELAP5/MOD3 and, in return, provides pressure and temperature of the containment to RELAP5/MOD3. Figure 3 shows the RELAP5/MOD3 break mass signatures provided to CONTAIN. Figures 4 and 5 show the pressure and

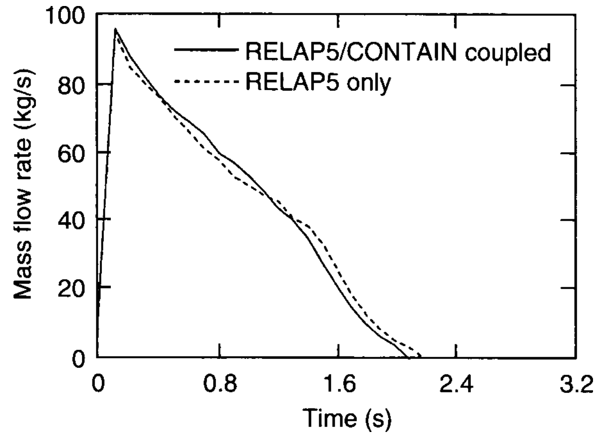

Fig. 3 Break mass flow rate for pressure vessel blowdown test case.

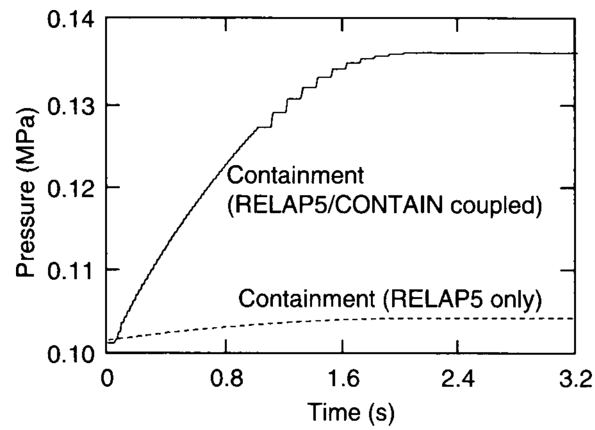

Fig. 4 Containment pressure during test case 1.

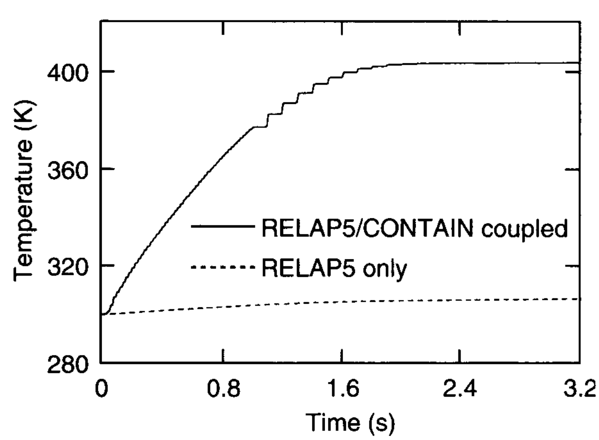

Fig. 5 Containment temperature during test case 1.

temperature response from CONTAIN. In contrast, the same simulation is plotted on these figures for a RELAP5/MOD3 simulation without the coupling with CONTAIN. The distinct differences observed from these results present an example of the importance this new feature can provide to best-estimate systems analysis. Containment pressure and temperature are noticeably higher for the coupled case, which is expected because the version of RELAP5/MOD3 used for this test does not normally incorporate a total energy equation (a special user option exists for convecting the total energy across a junction). Results from CONTAIN appear to advance in a stepwise fashion. This is evidence of the explicit coupling. At 1.0 s into the transient, the data exchange frequency between RELAP5/MOD3 and CONTAIN increases from 0.005 to 0.1 s. Clearly, this change has impact on the transient; however, under many circumstances this may be an acceptable approximation. The use of an explicit coupling will require that sensitivity studies be performed by the analyst to ensure that a proper communication frequency is used to adequately couple the problem. The parameters on the data exchange frequency input cards (2090001–99) represent the only coupling parameters that an analyst can change to investigate the sensitivity of calculations to the explicit coupling.

RELAP5/MOD3 Coupled with CONTAIN: SBWR Main Steam Line Break

The MSLB in the SBWR was a much more challenging test for the RELAP5/MOD3 and CONTAIN link. In this case, more than 50 variables were sent between the codes. Figure 6 shows a simple nodalization of the SBWR containment and indicates coupling locations. Table 1 identifies variables shared between the codes. A unique aspect of the SBWR containment is the passive containment cooling system (PCCS). The PCCS is responsible for long-term cooling of the containment during abnormal conditions. The containment atmosphere is driven into the PCCS through natural convection, and vapor is condensed while noncondensible gases are separated and driven into the suppression chamber. Because CONTAIN does not have a model for describing this component, RELAP5/MOD3 was used to mechanistically model the component the best way possible.

The most important lesson learned from using the coupling model has been the identification of what information should be passed between the codes. This became clear when performing the SBWR MSLB. The complex problem involved sending and receiving the multispecies (air, vapor, and liquid water) volume properties. Experience showed that, when sending information to RELAP5/MOD3 time-dependent volumes, the input model developer should ensure that the child process sends all the same variables described with the initial condition option. If this process is not followed, conflicting state properties can cause a code failure.

The MSLB transient was initiated by an instantaneous rupture of one steam line upstream of the main steam isolation valves (MSIVs), which resulted in a break that discharged to the drywell. Break flow from the reactor vessel was limited by restricting orifices in the steam nozzles. Break flow from the MSIV side was stopped almost immediately after break initiation as the MSIVs close quickly. For this demonstration, the transient calculation was terminated 60 s into the calculation.

Following the break, the reactor vessel pressure decreased rapidly, as shown in Fig. 7. In this figure and the following figures, results from the coupled RELAP5/CONTAIN calculation are compared with a calculation using RELAP5/MOD3 only. In the RELAP5/MOD3-only case, the input model includes the definition of all components that CONTAIN models for the RELAP5/CONTAIN case. In both the RELAP5/MOD3-only and the RELAP5/CONTAIN cases, the pressure signatures for the reactor coolant system were similar, as would be expected. Figure 8 shows the break flow from the steam line; for the RELAP5/CONTAIN case, this represents the boundary condition sent to CONTAIN from RELAP5/MOD3. As the vessel depressurized, liquid was pulled up the downcomer and into the broken steam line, as evidenced with the oscillation in the break flow.

Figures 9 to 12 show the pressure and temperature from the drywell and suppression chamber, respectively. One notable discrepancy is obvious; that is, the RELAP5/MOD3-only case predicts a blowout of the horizontal vents that exist between the drywell and suppression chamber. This is evident from the sharp increase and the following decrease in drywell pressure while the suppression chamber maintained a quasi-steady increase. The RELAP5/CONTAIN case predicted a gradual pressurization of both the drywell and suppression chamber without the blowout of the horizontal vents, which is a more plausible result because atmospheric pressure

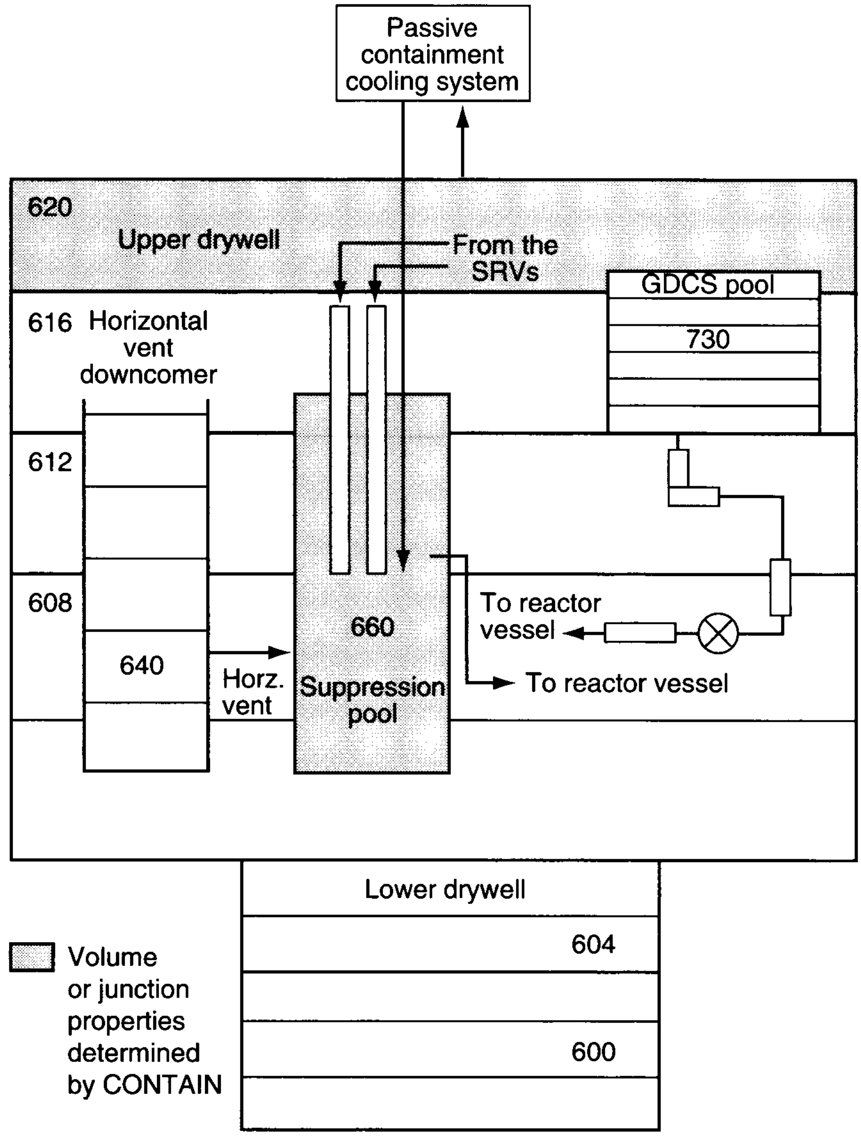

Fig. 6 SBWR containment RELAP5/MOD3 nodalization with coupling locations identified. [SRV is safety relief valve. GDCS is gravity-driven cooling system. The RELAP5 input file component descriptions are indexed by volume numbers, which are arbitrarily assigned.]

changes should be transmitted rapidly throughout the containment. The RELAP5/CONTAIN case shows that the magnitude of the pressure in the containment is higher than the RELAP5/MOD3-only case because RELAP5/MOD3 does not normally incorporate a total energy conservation equation across junctions.

As two-phase water from the break enters the drywell, the vapor displaces the air in the containment, and the two species are stratified with the lighter water vapor filling the topmost regions of the containment. The noncondensible quality in these regions quickly decreases, as shown in Fig. 13. The RELAP5/MOD3-only case does not decrease because RELAP5/MOD3 treats noncondensible gas as an entity mixed homogeneously with the vapor. The difference has a drastic effect on the performance of the PCCS, as shown in

Table 1 Coupling Variables for the SBWR MSLB

Phenomena	RELAP5/MOD3	CONTAIN
Break mass flow	X	
PCCS mass in-flow	X	
PCCS mass out-flow	X	
Break enthalpy flow	X	
PCCS enthalpy in-flow	X	
PCCS enthalpy out-flow	X	
Drywell pressure		X
Drywell temperature		X
Drywell void fraction		X
Drywell internal liquid internal energy		X
Drywell internal vapor/noncondensible internal energy		X
Noncondensible quality in drywell		X
Suppression chamber pressure		X
Suppression chamber temperature		X
Suppression chamber void fraction		X
Suppression chamber internal liquid internal energy		X
Suppression chamber internal vapor/noncondensible internal energy		X
Noncondensible quality in suppression chamber		X
Vacuum break mass flow		X

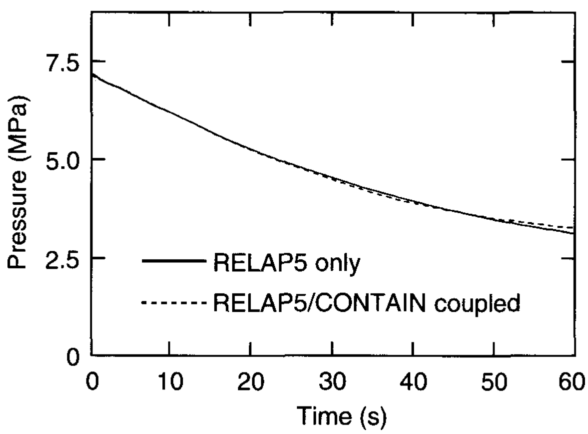

Fig. 7 Reactor pressure during the Simplified Boiling-Water Reactor main steam line break test case.

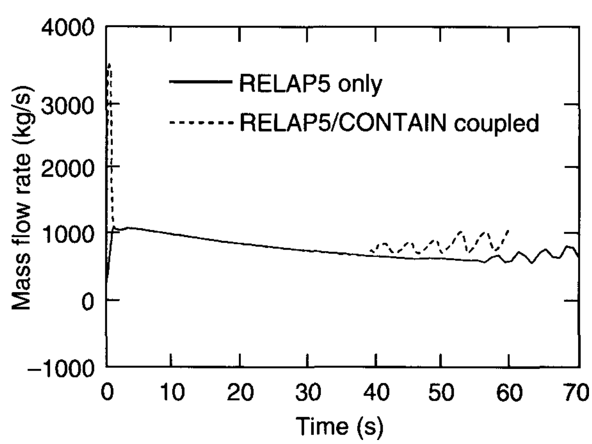

Fig. 8 Break flow during the Simplified Boiling-Water Reactor main steam line break test case.

Fig. 14. With the greater concentration of water vapor, the PCCS removes more than ten times the energy predicted by RELAP5/MOD3 only.

SUMMARY AND CONCLUSIONS

A new feature has been developed and implemented in RELAP5/MOD3 to allow the coupling of data between RELAP5/MOD3 and other codes. Specifically, the containment analysis code CONTAIN has been linked with RELAP5/MOD3. This feature uses the parallel process management and data transfer capabilities provided by the PVM software. An explicit couple method was used for this new model. An explicit couple discretely updates boundary conditions between codes rather than solving a combined solution matrix of the two processes, as required by a rigorous implicit couple model. Although the explicit model may be less accurate, it can be generally applied to many problems. Coupling with RELAP5/MOD3 is activated

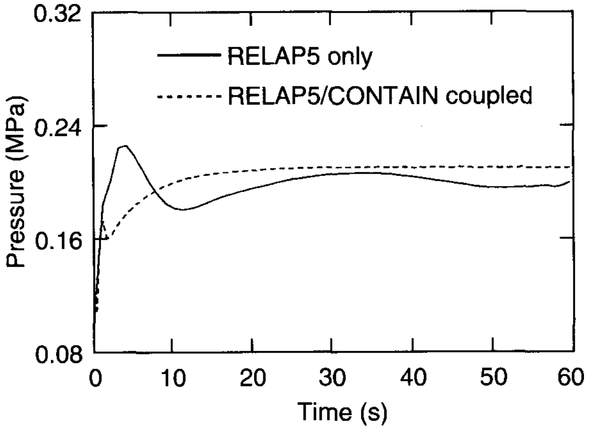

Fig. 9 Drywell pressure during the Simplified Boiling-Water Reactor main steam line break test case.

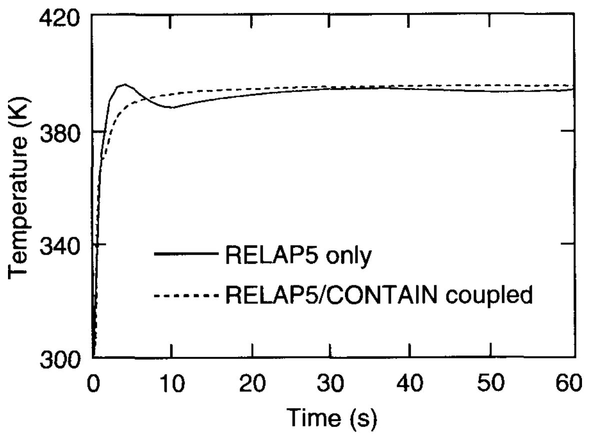

Fig. 10 Drywell temperature during the Simplified Boiling-Water Reactor main steam line break test case.

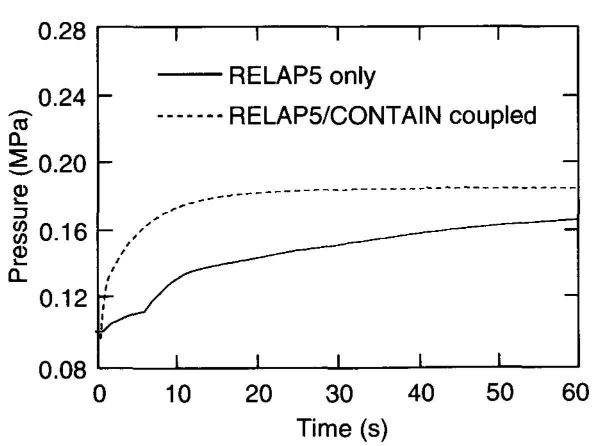

Fig. 11 Suppression chamber pressure during the Simplified Boiling-Water Reactor main steam line break test case.

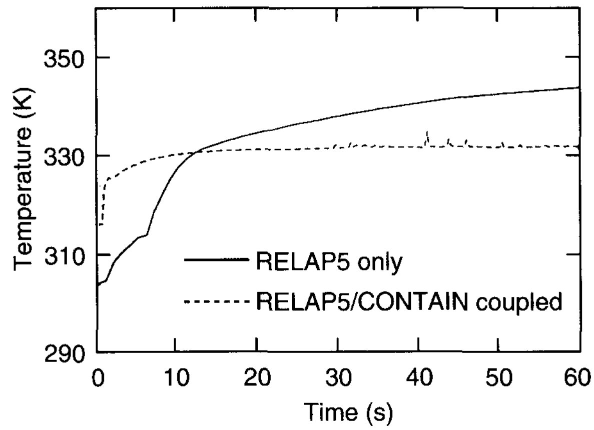

Fig. 12 Suppression chamber temperature during the Simplified Boiling-Water Reactor main steam line break test case.

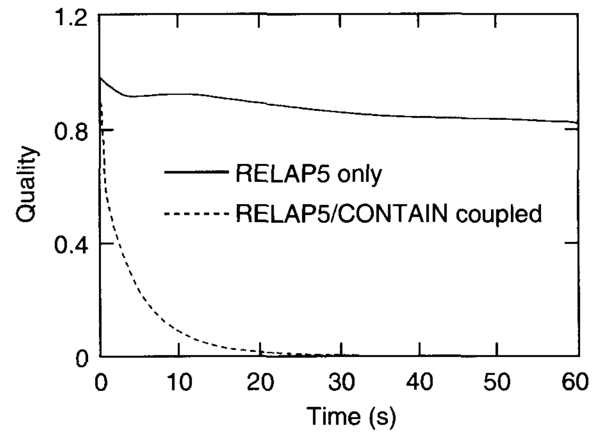

Fig. 13 Noncondensible quality during Simplified Boiling-Water Reactor main steam line break test case.

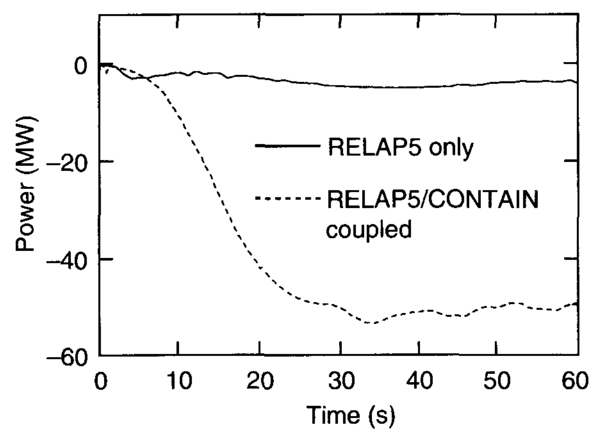

Fig. 14 PCCS power during Simplified Boiling-Water Reactor main steam line break test case.

by introducing new input into any standard RELAP5/MOD3 input model file that includes the name of the code to be coupled, a table of time-dependent data transmission frequencies, a table of variables to be sent to the coupled code (i.e., CONTAIN), and a table of variables to receive data from the coupled code. The infrastructure of this model has been designed to be as general as possible to allow the coupling of RELAP5/MOD3 with any code. Results with four test problems demonstrate the feasibility of the coupling model through the proper transmission, processing, and integration of data between RELAP5/MOD3 and other codes. The two test cases with RELAP5/MOD3 and CONTAIN discussed in this paper show that more accurate results can be obtained by applying the improved models in CONTAIN.

ACKNOWLEDGMENTS

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Missiles Caused by Severe Pressurized-Water Reactor Accidents

By R. Krieg[a]

Abstract: *For future pressurized-water reactors, which should be designed against core-meltdown accidents, missiles generated inside the containment present a severe problem for its integrity. The masses and geometries of the missiles, as well as their velocities, may vary to a great extent. Therefore a reliable proof of the containment integrity is very difficult.*

In this article the potential sources of missiles are discussed, and the conclusion was reached that the generation of heavy missiles must be prevented. Steam explosions must not damage the reactor vessel head. Thus fragments of the head cannot become missiles that endanger the containment shell. Furthermore, during a melt-through failure of the reactor vessel under high pressure, the resulting forces must not catapult the whole vessel against the containment shell. Only missiles caused by hydrogen explosions may be tolerable, but shielding structures that protect the containment shell may be required. Further investigations are necessary.

Finally, measures are described showing that the generation of heavy missiles can indeed be prevented. Investigations are currently being carried out that will confirm the strength of the reactor vessel head. In addition, a device for retaining the fragments of a failing reactor vessel is discussed.

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THE SAFETY CONCEPT AND THE ROLE OF MISSILE IMPACT

For future pressurized-water reactors, it is not sufficient to show that core-meltdown accidents are very unlikely. In Germany, for instance, it will be demanded (as a result of a recent federal law) that, in addition, core meltdown accidents must not be able to impair the reactor containment so that severe consequences outside the plant can be excluded.[1–5] Consequently the evacuation of people and a temporary loss of land need not be discussed any longer, and the understanding and acceptance of the low probability for a core-meltdown accident, which has turned out to be a difficult hurdle for some people, is no longer an indispensable element of the safety concept. In other words, *the probabilistic approach is replaced by a deterministic point of view.*

For experts, extended safety requirements may provide just a further reduction of risk that is not absolutely necessary when risks of other human activities are taken into account. Nevertheless, the requirements are reasonable if one considers that the number of nuclear plants will increase and that the costs expected for the discussed improvements are moderate.

For concerned people, however, who hardly have the opportunity to check calculated probabilities and risk assessments in detail, the additional requirement may be of great help in appreciating that nuclear reactors do not present an undue safety problem. Also, it should be kept in mind that it is not only scientists who finally decide about the future of nuclear power.[6,7]

For compliance with this general aim, the worst-case containment loadings caused by core-meltdown accidents must be considered. These loadings include the impact of missiles stemming from (1) steam explosions, (2) a melt-through failure of the lower vessel head under high internal pressure, and (3) hydrogen explosions that may accompany the accident. As already emphasized some time ago,[8] the missile problem requires special attention because the masses, shapes, and velocities of missiles vary to such an extent that the specification of worst cases is quite difficult (for example, the mass of the whole pressure vessel, which may become a missile, amounts to some hundred thousand kilograms, whereas the masses of structural fragments that may hit the containment shell could be on the order of only 10 kg).

MISSILES CAUSED BY STEAM EXPLOSIONS

During a core-meltdown accident, large masses of molten fuel and other core material may fall into a water pool remaining in the lower head of the reactor vessel. If the lower head has already been melted through, the fuel may fall into water collected in the reactor vessel cavern. In the first case, an in-vessel steam explosion may occur, and in the second case, an out-of-vessel steam explosion may occur, which would accelerate molten-fuel slugs upward against the reactor vessel head.[9,10] If the head or its bolts fail, heavy fragments may be hurled against the containment shell (Fig. 1).

For 1300-MW pressurized-water reactors, the masses of the molten-fuel slugs have been estimated up to 80 000 kg. Their velocities are not known, but for safety analyses, figures around 150 m/s are under discussion.[11,12] Then the resulting momentum is around 12 MN and the kinetic energy is about 1000 MJ. The masses, velocities, and energies of missiles caused by a failing reactor vessel head could reach the same orders of magnitude. The transfer of kinetic energy into potential energy before the containment shell is hit can be neglected for the high velocities discussed here.

For comparison, the energy needed to deform 1 m^2 of a steel containment shell 40 mm thick such that the average strain reaches 20% is only about 4 MJ. For a piece of shell undergoing such a loading, significant leakages are likely. Therefore it can be concluded that missiles caused by a steam explosion damaging the reactor vessel would strongly endanger the containment integrity.

Even the energy dissipation by protective structures located inside the containment within a certain distance from the containment shell would hardly be sufficient. Assume, for instance, that the protective structures consist of concrete beams with width b and thickness h. Under missile loading, the beams would undergo bending exceeding the yield limit (σ_y) and form a plastic hinge that would be able to transfer a bending moment of

$$M = \frac{3}{2}\ \sigma_y\ \frac{bh^2}{6} \tag{1}$$

Assume further that the plastic hinge allows a bending angle (α) before the beam collapses and the bending moment vanishes. Then the dissipated energy is roughly

$$E = M\alpha \tag{2}$$

For heavy beams with b = 5 m and h = 2 m, with an average yield stress (σ_y) of 60 MPa, and with a relatively high bending angle (α) of 0.1 (5.7°), the dissipated energy amounts to

$$E = 30\ \mathrm{MJ} \tag{3}$$

This is much smaller than the kinetic energy of the missiles discussed previously (Fig. 2).

If, however, the reactor vessel head does not fail, the momentum of the molten-fuel slug is transferred to the whole vessel. Because of its large mass, its velocity is quite moderate. Therefore the vessel does not reach the dome of the containment shell.

UPWARD ACCELERATION OF THE WHOLE REACTOR VESSEL CAUSED BY A MELT-THROUGH FAILURE OF ITS LOWER HEAD UNDER HIGH INTERNAL PRESSURE

Depending on the initial events, the core-meltdown accident may start with the reactor vessel loaded by the operating pressure of about 160 bars. If no measures are taken to reduce this pressure, the melt-through failure of the reactor vessel lower head may cause very strong dynamic forces and thus catapult the whole reactor

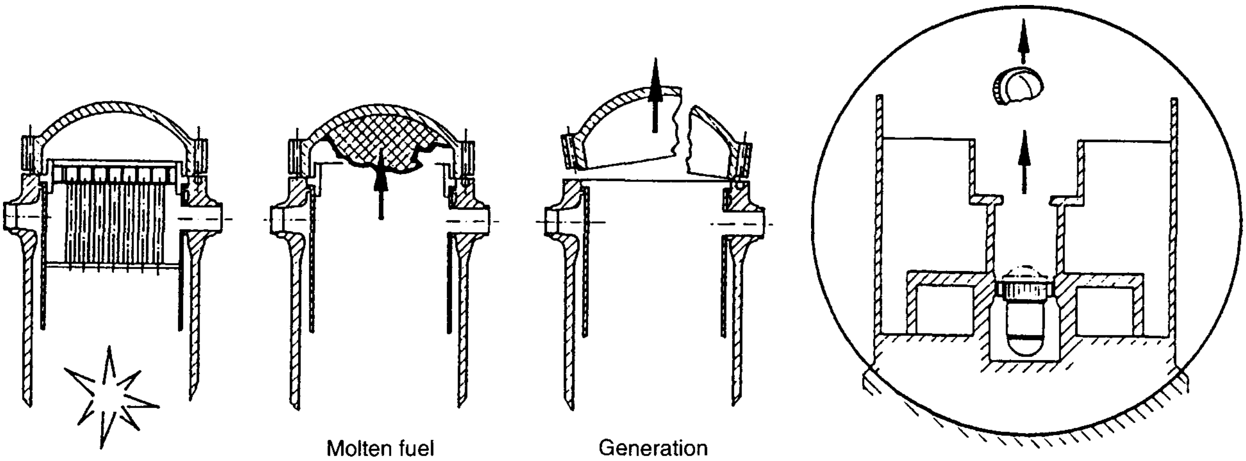

Fig. 1 Heavy fragments of the pressure vessel might be hurled against the containment shell as a consequence of a postulated steam explosion.

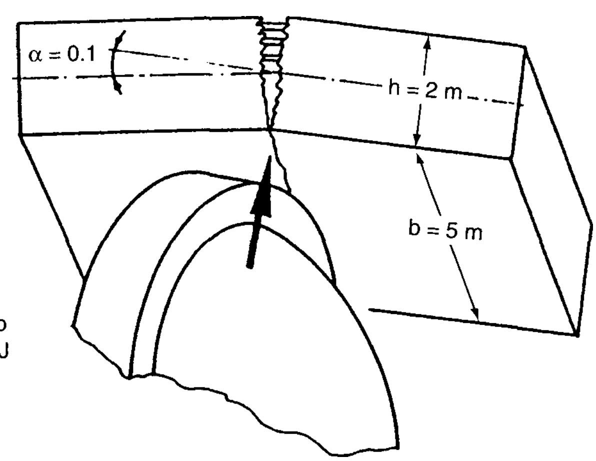

Fig. 2 The energy that can be dissipated by the protective structure is much smaller than the energy of missiles caused by a steam explosion [α = 0.1 (5.7°) is the bending angle of the beam].

vessel upward like a rocket (Fig. 3). According to actual knowledge, it cannot be ruled out that during this process the lower head will be completely torn off. For this case, maximum dynamic forces up to 300 MN occur.[13] The time history of these forces $[F(t)]$ is shown in Fig. 4. The maximum force exceeds the strength of the actual reactor vessel clampings considerably. If no adequate design changes are made, the clamping can be neglected and the momentum of the upward-moving vessel can be assessed as

$$I = \int F(t)\,\mathrm{d}t \tag{4}$$

With the use of the time history of Fig. 4, the following can be obtained:

$$I = 20\ \mathrm{MN} \tag{5}$$

For a reactor vessel mass of 500 000 kg, the upward velocity is

$$v = 40\ \mathrm{m/s} \tag{6}$$

and the kinetic energy is

$$E_1 = 400\ \mathrm{MJ} \tag{7}$$

In a vertical distance of about 40 m, the reactor vessel will hit the containment shell. After subtraction of the potential energy, which will be consumed before the containment shell is reached, the kinetic energy left is reduced to

$$E_2 = 200\ \mathrm{MJ} \tag{8}$$

This is again much more than the energy needed to damage the containment shell. It is also more than the energy that can be dissipated by the heavy protective structures discussed in the preceding section.

MISSILES CAUSED BY HYDROGEN EXPLOSION

As a consequence of a core-meltdown accident, hydrogen may be released into the containment

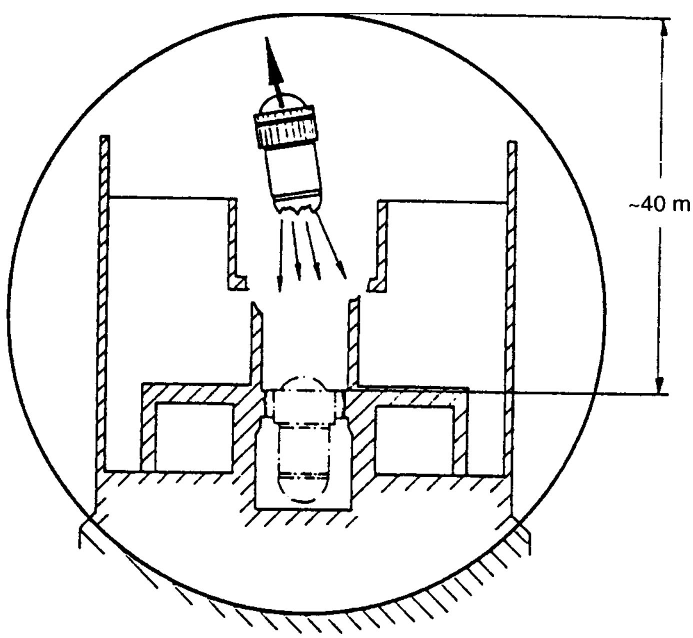

Fig. 3 The whole reactor vessel might be hurled against the containment shell as a consequence of a postulated melt-through failure under high internal pressure.

atmosphere. If no measures are taken to reduce the hydrogen accumulation, it may detonate and thus propagate strong pressure waves through the containment atmosphere.[14] These waves will also pass structural elements surrounded by the containment atmosphere. Consequently, during this process pressure differences will act on these elements. In addition, the flowing gas behind the wave front will exert drag forces on the elements. Therefore structural elements that are not properly fixed will be accelerated.[8] In this way they could become missiles (Fig. 5).

The driving forces caused by the propagating waves act for only very short times, however. Because the velocity of the waves is always larger than the velocity of the flowing gas, which again is larger than the velocity of the structural elements, the waves will be reflected at the containment wall and will come back before the structural elements have reached the containment wall. The acceleration of the structural elements will then be reversed. Because of the three-dimensional effects of the processes, the superposition of acceleration and deceleration may not cancel the movement of

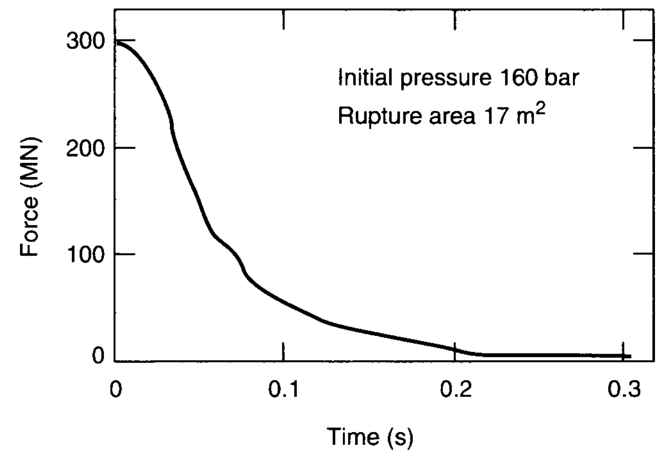

Fig. 4 Force F acting at the reactor vessel after a melt-through failure under high internal pressure. See Ref. 13; figure used with permission.

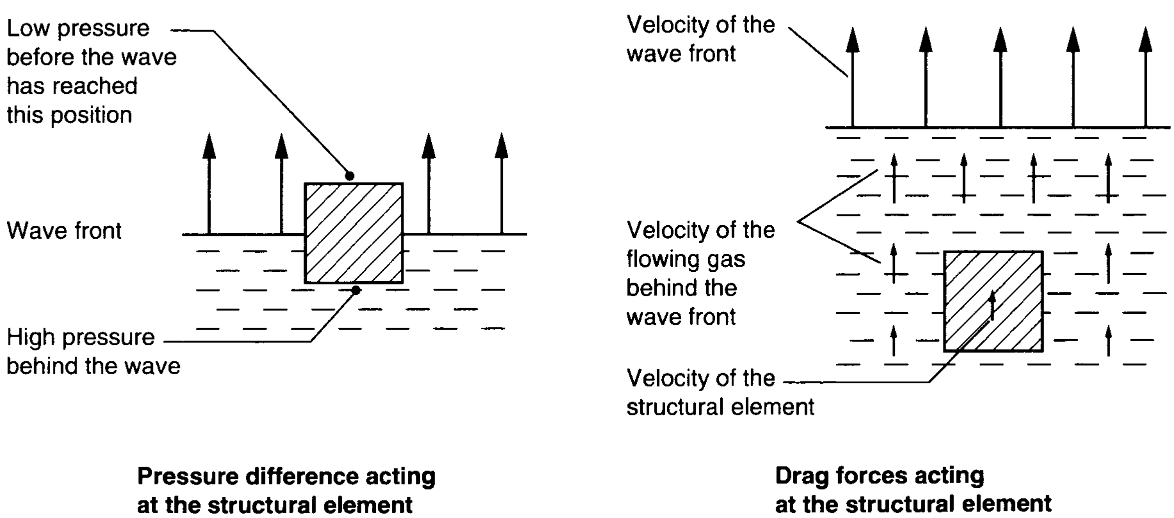

Fig. 5 Structural elements that are not properly fixed may be accelerated by passing pressure waves caused by a postulated hydrogen explosion.

the elements completely, but the velocity should be reduced considerably before the elements hit the containment wall. In conclusion, the velocity of missiles caused by hydrogen explosions is expected to be moderate. In contrast, the overpressure from the hydrogen explosion will be a more severe problem for the containment shell, but that topic is not the subject of this paper.

The missile problem may be more critical if the hydrogen explosion occurs in a compartment with small openings. Then the pressures may act for a longer time, and if a compartment breaks, fragments of the walls may be accelerated to higher velocities.

CONSEQUENCES FOR THE PLANT DESIGN

As stated earlier, the protection of the containment shell against missiles caused by steam explosions breaking the reactor vessel head would be very difficult. Concrete shielding 2 m thick inside the containment shell would hardly be sufficient. Therefore the vessel head must be able to withstand the dynamic loading caused by a steam explosion. Then missiles caused by steam explosion would no longer challenge the containment integrity. Investigations of the load-carrying capacity of the reactor vessel head will be discussed in the following section.

Also as stated earlier, the protection against the impact of the whole reactor vessel accelerated by a melt-through failure under high internal pressure would be almost impossible. Therefore one of the following is required: (1) the pressure in the reactor vessel must be sufficiently reduced before the melt-through failure occurs (this solution would require only minor changes, but the necessity of active measures may be criticized), or (2) the hole in the reactor vessel caused by the melt-through failure can be shown to be always much smaller than the cross section of the vessel (such a proof would be quite difficult, even after appropriate changes of the geometry of the vessel), or (3) the clamping of the vessel must be improved significantly (this solution would require much stronger and space-consuming designs), or (4) the accelerated reactor vessel must be caught by a missile-retention device.

Strictly speaking, these measures do not have to prevent upward movement of the reactor vessel completely. It would be sufficient if the initial kinetic energy of the vessel were smaller than the consumption of potential energy before the vessel reaches the containment shell. This would be the case if the initial velocity of the vessel is smaller than about 25 m/s. Therefore some of the preceding measures that are difficult to realize when the reactor vessel must be kept at its place could be quite suitable when only its initial velocity must be reduced (for instance, the improvement of the vessel clamping should also be evaluated from this point of view). The same is true for the missile-retention device, which will be discussed in more detail.

If the preceding section is taken into account, protection of the containment shell against missiles with moderate velocities stemming from hydrogen explosions seems to be possible;[8] however, more detailed investigations are needed to evaluate the velocity that can be reached by such missiles. Also, details of the impact process with the containment shell and the expected leakage are not sufficiently known. To avoid missiles with higher velocities, the inner containment structures must be such that the pressurization and the collapse of individual compartments cannot occur. Aside from the containment shell, the sensitivity of the particular containment walls under missile loading also must be studied. Of course, the dynamic interaction with shielding structures must be included.

INVESTIGATION OF THE SLUG IMPACT STRENGTH OF THE REACTOR VESSEL HEAD

Assessments carried out recently suggest that rather strong molten-fuel slug impacts can be tolerated by the vessel head. For slug masses up to 80 000 kg, tolerable velocities between 150 and 210 m/s are mentioned.[11,12] This means that steam explosions causing such impacts cannot be a source for missiles endangering the containment shell.

Reliable proofs of the slug impact strength of the reactor vessel head are quite difficult. If the upper internal structures underneath the vessel head are neglected, computational models can be applied to describe the impact problem. Exploratory computations show that some basic phenomena of the liquid-structure impact are not fully understood; different computational models yield different results. Therefore experiments would be necessary to clear up this problem. Whatever the outcome, the results will depend very much on the assumed slug shape. If the slug fits well into the reactor vessel head, the load peaks will be very high. This possibility presents a problem because in safety investigations such extreme cases must be considered.

If the upper internal structures are included, the liquid-structure impact process is much more difficult. As indicated by several assessments, the internal structures will be heavily damaged during the impact.[9,11,12] Under this condition, the development of appropriate computational models is almost impossible, and their results will be very questionable. Conversely, however, the interaction of the liquid slug with the failing upper internal structures is quite important. The slug shape is not expected to influence the results very much, which is essential for safety investigations. Slug shapes that would fit into the vessel head and therefore cause very strong load peaks will be disturbed during penetration through the upper internal structures. In addition, the interaction with these structures will dissipate kinetic energy and smooth the impact process; i.e., it will increase the duration (Δt) of the impact. Consequently the interaction with the upper internal structures will lead to an attenuation of the loading. It will increase the slug momentum (I) that can be carried by the vessel head:

$$I = \int_{\Delta t} F(t)\, \mathrm{d}t \tag{9}$$

Note that $F(t)$ is the impact forces vs. time (t) and that the maximum impact force is given by the strength of the bolts, for instance. The problem is illustrated in Fig. 6.

To avoid computational difficulties, the impact problem will be investigated by the model experiments BERDA (Beanspruchung des Reaktordruckbehälters bei einer Dampfexplosion), where the impact process will be simulated in a smaller scale such that the expenses are acceptable. For similarity between full and small scale, essential dimensionless quantities must be matched. They can be identified with the use of the relevant basic equations describing the problem.[12,15]

The test facility for the model experiments BERDA is shown in Fig. 7. The structural model is scaled down by a factor of 10. It consists only of the upper part of the reactor pressure vessel, including the vessel head with its bolts, and the upper internal structures with the grid plate, the guide tubes and support columns, the upper support grid, and the upper part of the core barrel. A liquid of the same density will be used to simulate the molten fuel slug. It will be accelerated upward to a predefined velocity with the use of a pneumatic drive mechanism. During the acceleration phase, the liquid is contained in a crucible that is able to withstand the acceleration forces. Before the liquid reaches the head, the crucible will be decelerated by a crash material while the upward movement of the liquid slug continues until penetration into the upper internal structures and impact at the vessel head occurs. The maximum mass of the slug is 80 kg, which corresponds to 80 000 kg in full scale. The maximum slug velocity is 130 m/s, which corresponds to about the same velocity in full scale.

Care has been taken for sufficient instrumentation. The slug velocity and the slug shape will be measured before the slug impact. Resulting pressures, forces, strains, and accelerations of the structures will be recorded during the test as a function of time. Permanent deformations and the amount of fracturing can be determined in great detail after the test.

The results can be directly transferred to full scale (for instance, the strains in the model are the same as the strains in full scale). In particular, slug velocities that the model can withstand will also be tolerable for the real pressure vessel.

DESCRIPTION OF A CAGE-TYPE MISSILE-RETENTION DEVICE FOR A BURSTING REACTOR VESSEL

According to assessments by Eibl, an improvement of the vessel clamping should be possible such that even under worst conditions the whole vessel will not be able to hit the containment shell.[16] Nevertheless, several proposals have been made as to how to protect the containment shell against a pressure vessel catapulted upward.

About 20 years ago a burst-protection device was discussed intensively in Germany. At that time the aim was to keep the fragments in their place and to avoid large cracks so that the cooling process could be maintained. It turned out that such an ambitious device has other severe drawbacks. Therefore it was finally rejected.[17] In the meantime, other devices that retain the fragments of the vessel in the reactor cavity only have been discussed. Such devices are sufficient to protect the containment shell against missile impact and have fewer drawbacks. As some studies show, these proposals seem to be feasible with reasonable effort[1,18,19] [for example, a cage-type missile-retention device described in Ref. 19 and shown in Fig. 8(a) will be discussed in some detail]. It is designed for protection against sudden vessel failure caused by cracks propagating in both circumferential and axial directions. Of course, if such a device were introduced, the investigation of the slug impact strength at the vessel head would be of lower priority.

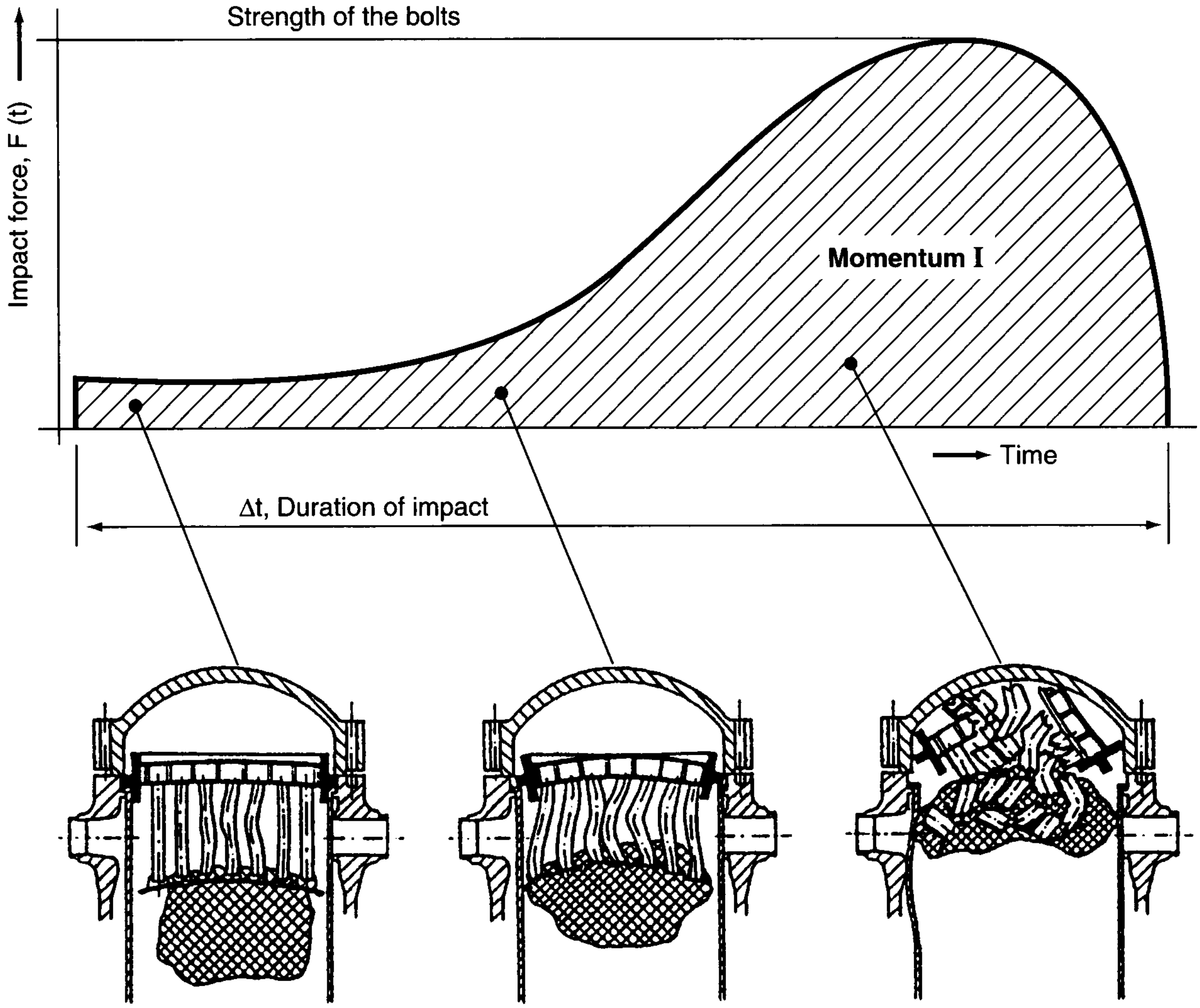

Fig. 6 Force transferred during the slug interaction with the upper internal structures and the impact with the head. The curve shows only the qualitative behavior.

A gap of 0.5 m exists between the reactor vessel and any surrounding structure, so access to the reactor vessel is possible. Upon failure of the reactor vessel under high pressure, the fragments are accelerated across this gap to high velocities before they impact the missile-retention device. This consists of individual rings and axial bars, which are made of a high-strength ductile steel and which are designed to undergo considerable uniaxial plastic elongations under the impact. Thus the high kinetic energy of the vessel fragments will be dissipated in these elements without threatening the containment shell.

The innermost concrete structure fixes the rings and carries their dead weight. (Alternatively, the rings could be attached directly to the axial bars.) Appropriate radial openings in the concrete structure along the circumference will distribute the escaping steam more symmetrically. For refueling, the nuts of the bars must be detached before removing the upper traverse. Then the refueling conditions are similar to those in present plants.

The elements of the missile-retention device form a closed system. Major unbalanced forces caused by jets escaping from the failing reactor vessel, which might catapult the system away, cannot occur. Therefore heavy clamping of the missile-retention device is not required.

Attention must be paid to the strain distributions along the rings and axial bars. Because of wave

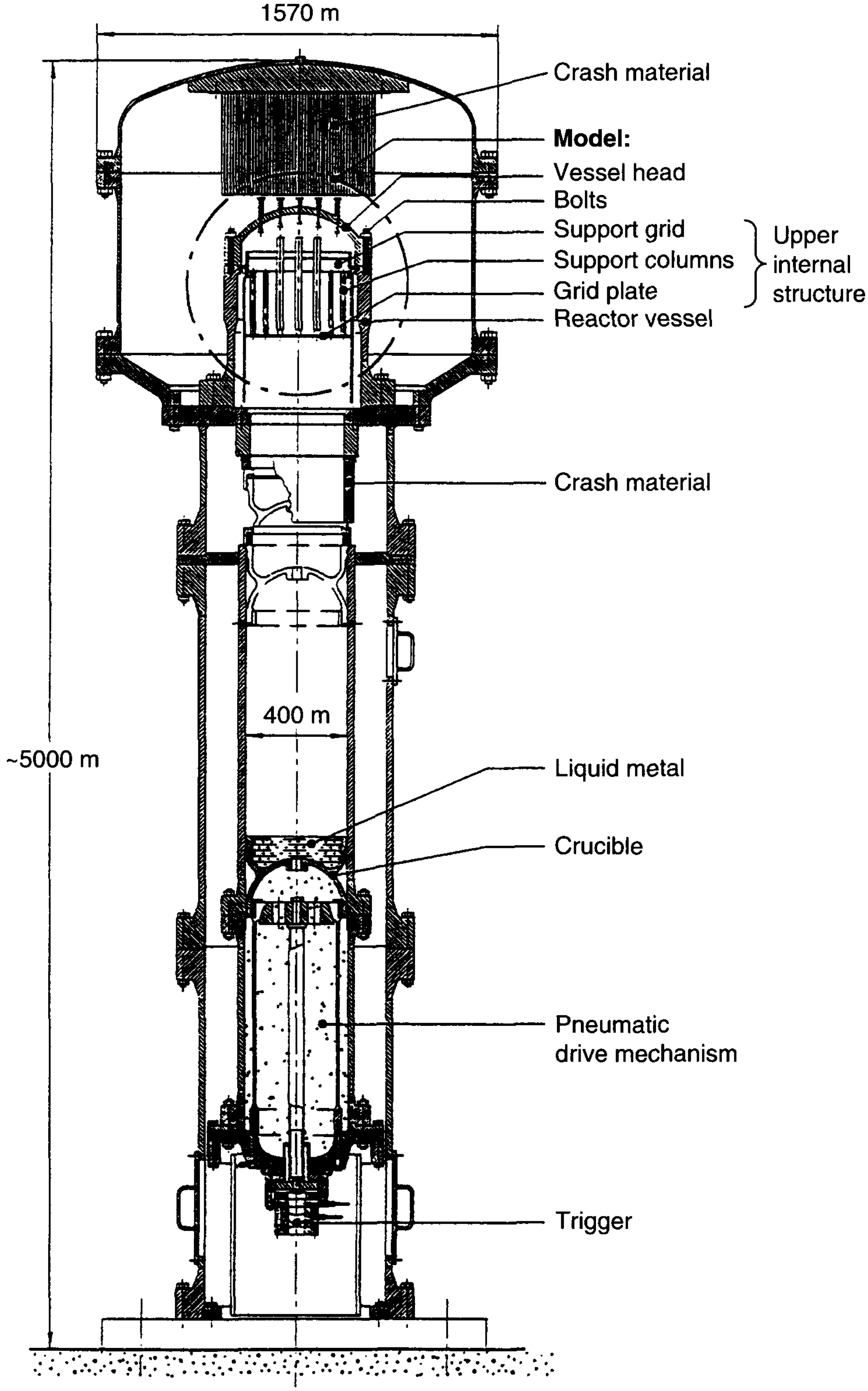

Fig. 7 Test facility for the model experiments BERDA; the model is scaled down by 1:10.

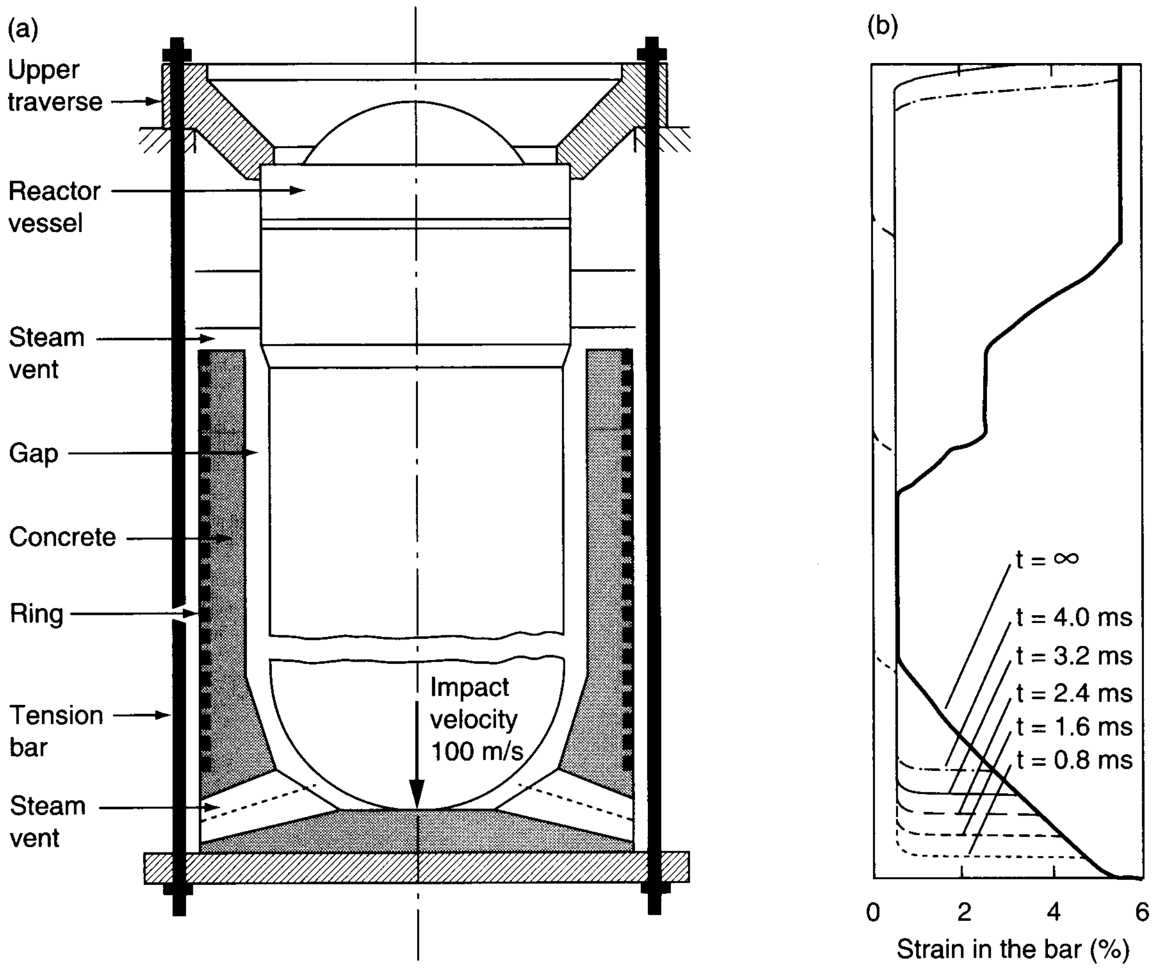

Fig. 8 Illustration showing the proposed missile retention device (a) and distribution along the axial bar for different times (t) after an impact of the broken lower head (b).

propagation effects, the distributions can be very irregular. Figure 8(b) shows the distribution along the axial bar for different times (t) after an impact of the broken lower head. In the elastic region up to a strain of about 0.5%, the wave propagation is fast. In the plastic region (i.e., for strains above 0.5%), the propagation is considerably slower. The accumulated strain distribution is shown by the bold curve. It varies between 0.5% and almost 6%.

CONCLUSIONS

The containment shell must not be hit by heavy missiles caused during a serious steam explosion or a melt-through failure of the pressure vessel under high internal pressure. Protective structures in front of the containment shell would not be sufficient. Research activities are under way or have been carried through to show that such missiles will not be generated or can be retained by adequate devices. Of course, detailed stress analyses for special devices must be provided later during the design phase; however, smaller missiles that may be caused during a hydrogen explosion may be tolerable. The need exists for detailed investigations of the velocities that such missiles may reach and of the impact at the containment shell, which would yield information about the resulting leakage.

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Design Features

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Validation of COMMIX with Westinghouse AP-600 PCCS Test Data

By J. G. Sun,[a] T. H. Chien,[a] J. Ding,[b] and W. T. Sha[a]

Abstract: *Small-scale test data for the Westinghouse AP-600 Passive Containment Cooling System (PCCS) have been used to validate the COMMIX computer code. So that the performance of the PCCS can be evaluated, two transient liquid-film tracking models have been developed and implemented in the COMMIX code. A set of heat-transfer models and a mass transfer model based on heat and mass transfer analogy were used for the analysis of the AP-600 PCCS. The flow of the air stream in the annulus is a highly turbulent forced convection, and the flow of the air–steam mixture in the containment vessel is a mixed convection. Accordingly, a turbulent-forced-convection heat-transfer model is used on the outside of the steel containment vessel wall and a mixed-convection heat-transfer model is used on the inside of the steel containment vessel wall. The results from the COMMIX calculations are compared with the experimental data from Westinghouse PCCS small-scale tests for average wall heat flux, evaporation rate, containment vessel pressure, and vessel wall temperature and heat flux distributions; agreement is good. The COMMIX calculations also provide detailed distributions of velocity, temperature, and steam and air concentrations.*

The AP-600, an advanced pressurized-water reactor, uses a passive containment cooling system (PCCS) to remove heat released inside the containment vessel following postulated design-basis accidents (DBAs) such as a main-steam-line break or loss-of-coolant accident. During a DBA, heat released to the interior of the steel containment vessel is removed by evaporation of a continuously flowing thin liquid film on the outside surface of the vessel, which lowers the temperature of the steel vessel wall so that steam condenses on its inside surface. Consequently the pressure inside the containment vessel is lowered. The external liquid film is formed by flooding water at the top of the ellipsoidal dome. Evaporation of the falling liquid film is enhanced by buoyancy-driven flows of moist air in an annular space outside the steel containment vessel.

For PCCS performance, it is necessary to predict both the evaporating film on the outside surface of the vessel and the condensate film on its inside. To this end, two liquid-film tracking models for time-dependent flows (a simplified model and a comprehensive model) have been developed and implemented in the COMMIX code.[1–5] COMMIX is a general-purpose, time-dependent, multidimensional computer code for thermal–hydraulic analysis of single-component or multicomponent engineering systems. It solves a system of conservation equations of continuities for up to six species, mixture momentum, mixture energy, and a k-ε two-equation turbulence model. A unique feature of the COMMIX code is its porous-media formulation,[6] which represents the first unified approach to thermal–hydraulic analysis. The tracking thermal–hydraulic models[7] compute the transient liquid-film thickness, velocity, and temperature on both sides of the steel containment vessel. Coupled with the liquid-film tracking models, pertinent heat and mass transfer models have been developed and implemented. Heat and mass transfer models were assessed by incorporating them into COMMIX and then comparing the computed results

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with the six experimental data sets for average wall heat flux, evaporation rate, vessel pressure, and vessel wall temperature and heat flux distributions obtained from the Westinghouse PCCS small-scale tests.

LIQUID-FILM TRACKING MODELS

Because the performance of a reactor containment system following an accident is highly transient, the decision was made to develop liquid-film tracking models suitable for time-dependent flows. The models are particularly relevant for the condensate inside the containment vessel. The containment vessel is a vertical cylindrical shell capped at both top and bottom by an ellipsoidal dome.

The dynamic equations for steady, axisymmetrical, boundary-layer flow over a body of revolution were first given by Boltz[8] and are most conveniently expressed in orthogonal curvilinear coordinates. The coordinate x will be denoted as the distance measured along a meridian from the stagnation point on the upper dome, and the coordinate y will be denoted as normal to x and pointing inward for analysis of condensate film flow on the inside surface and outward for analysis of evaporating water film on the exterior. The body contour is specified by the radius $[r(x)]$ of the section perpendicular to the axis of symmetry. The velocity components along the x and y directions are denoted by u and v, respectively. The gravitational acceleration vector ($\mathbf{g}$) is colinear with the axis of rotational symmetry, and the angle between $\mathbf{g}$ and x is β. The condensate flows along the undersurface of the dome and then down the vertical wall of the cylinder shell, mainly because of gravity, and is resisted by fluid viscosity. The pressure gradient in the streamwise direction is assumed to be small relative to the gravitational and viscous forces. Because the temperature difference across the film is small, typically about 2 °C, all properties can be considered constant in the derivation. The same can be said for the flow of the evaporating water film on the outside. The condensate flow rate increases with x with a continuous increase in resistance to heat transfer along the vertical cylindrical wall. The water film on the outside evaporates such that its flow rate decreases with x with a corresponding change in heat-transfer coefficient.

The time-dependent conservation equations for the film follow.

Mass:

$$\frac{\partial(ru)}{\partial x} + \frac{\partial(rv)}{\partial y} = 0 \tag{1}$$

Momentum:

$$\rho\left(\frac{\partial u}{\partial t} + u\frac{\partial u}{\partial x} + v\frac{\partial u}{\partial y}\right) = \mu\frac{\partial^2 u}{\partial y^2} + \rho g\cos\beta \tag{2}$$

Energy:

$$\rho C_p\left(\frac{\partial T}{\partial t} + u\frac{\partial T}{\partial x} + v\frac{\partial T}{\partial y}\right) = k\frac{\partial^2 T}{\partial y^2} \tag{3}$$

In Eq. 2, μ and ρ are the viscosity and density of the liquid film, respectively. In Eq. 3, T is the temperature and k and C_p are thermal conductivity and specific heat of the liquid, respectively.

The boundary conditions for the film are

At the wall, $y = 0$:

$$u = v = 0 \tag{4a}$$

$$T = T_w \tag{4b}$$

At the liquid–vapor interface, $y = \delta(t,x)$:

$$\frac{\partial u}{\partial y} = 0 \tag{5a}$$

$$\rho\frac{\partial \delta}{\partial t} - \rho v + \rho u\frac{\partial \delta}{\partial x} = m_c \tag{5b}$$

$$k\frac{\partial T}{\partial y} = m_c h_{fg} + h_c(T_b - T_i) \tag{5c}$$

In Eq. 4b, T_w is the local temperature on the wall surface. The condition $\partial u/\partial y = 0$ implies that interfacial shear is negligible. Equation 5b is the interfacial kinetic relation, in which m_c is the condensate mass flux. Equation 5c is the interfacial heat balance relation, in which h_{fg} is the enthalpy of condensation. The second term on the right-hand side of Eq. 5c accounts for the convective heat flux resulting from the bulk motion of the vapor–air mixture in the containment vessel. It is expressed in terms of convective heat-transfer coefficient (h_c) and a temperature difference ($T_b - T_i$); here T_b is the bulk temperature of the local vapor–air mixture and T_i is the local interfacial temperature. The initial condition is simply

$$\delta(t = 0, x) = 0 \tag{6}$$

Two liquid-film tracking models have been developed from Eqs. 1 to 6, a simplified model and a comprehensive model. The final formulations of the models were expressed in finite-difference forms, so they can be readily implemented in the COMMIX code. In the simplified model, the inertia terms in the momentum equation were ignored to obtain an explicit expression of the velocity field in the film. In the comprehensive model, the inertia terms were retained in the momentum equation, and a parabolic velocity distribution was assumed across the film thickness. In both models, linear temperature distributions across the film thickness were assumed in the energy equation. Both models have been used to analyze AP-600 PCCS small-scale integral test data.[7] The two models gave essentially the same result.

HEAT AND MASS TRANSFER MODELS

Solutions of the energy equation would require information on the local convective heat-transfer coefficient between the bulk of the vapor–air flow and the gas–liquid interface or the surface of the vessel wall. Judging from the large height of the system, one would expect that buoyancy plays an important role. The heat and mass transfer models used in the CONTAIN code[9] are described in the next two sections, respectively, and are implemented in the COMMIX code. A preliminary turbulent-mixed-convection (TMC) model is then described.

Heat-Transfer Models in the Contain Code

Several correlations for the Nusselt number [Nu (= h_cL/k)] were used in CONTAIN. The characteristic length (L) in Nu is chosen on the basis of the problem under investigation. The convective heat-transfer coefficient (h_c) is then used in conjunction with a temperature difference (ΔT) to calculate the convective heat flux to or from the vapor–gas mixture–liquid interface or the dry wall surface. Thus the heat flux is

$$q = h_c(T_b - T_i) \tag{7}$$

where T_b is the fluid bulk temperature and T_i is either the vapor–gas mixture–liquid-interface temperature or the dry wall temperature. The correlations follow.

For laminar natural convection:

$$h_1 = \frac{k}{L}\left[0.27(\mathrm{Gr\,Pr})^{1/4}\right] \tag{8a}$$

For turbulent natural convection:

$$h_2 = \frac{k}{L}\left[0.14(\mathrm{Gr\,Pr})^{1/3}\right] \tag{8b}$$

For turbulent forced convection:

$$h_3 = \frac{k}{L}\left[0.037\,\mathrm{Re}^{4/5}\,\mathrm{Pr}^{1/3}\right] \tag{8c}$$

In the preceding equations, Pr is the Prandtl number, and the characteristic length in the Grashof number (Gr) and in the Reynolds number (Re) is also L.

Mass Transfer Models

A small quantity of noncondensable gas in vapor would have a great effect on the condensation rate. The noncondensable gas will accumulate at the vapor–liquid interface and thus increase the partial pressure of the gas at the interface with a simultaneous reduction of the partial pressure of the vapor because the sum of the two remains constant. The mass transport of vapor–gas mixture consists of vapor diffusion toward the interface and gas diffusion away from the interface in addition to the bulk convection of the mixture. The interface temperature (T_i) is taken to be the saturation temperature corresponding to the vapor partial pressure (p_{vi}) at the interface. If p_{vb} denotes the partial pressure of the vapor in the bulk mixture, then the mass flux of vapor at the interface is

$$m_c = K_g \rho \ln \frac{p_T - p_{vi}}{p_T - p_{vb}} \tag{9}$$

where K_g is the mass transfer coefficient (having the dimension of velocity), ρ is the mixture density, and p_T is the total system pressure. If one stipulates that heat and mass transfer processes are analogous, then

$$K_g = \frac{h_c}{C_p \rho}\left(\frac{\mathrm{Pr}}{\mathrm{Sc}}\right)^{2/3} \tag{10}$$

where h_c is the mixed-convection heat-transfer coefficient; C_p is the specific heat of the mixture; and Sc is the Schmidt number ν/D, with D the mass diffusivity.

Turbulent-Mixed-Convection Model

In mixed convection, the relative importance of the natural- and forced-convection components can be measured by the ratio Gr_x/Re_x^2, where Gr_x is the local Grashof number and Re_x is the local Reynolds number. A commonly used rule is that when $Gr_x/Re_x^2 = O(1)$, the two components are of equal importance. The forced-convection effect will dominate when $Gr_x/Re_x^2 \ll 1$, and the natural-convection effect will dominate when $Gr_x/Re_x^2 \gg 1$. By examining the ratio along the vessel wall surface under the test conditions, Chien et al.[10] found that the ratio is O(1) on more than half of the total surface area. The ratio is small near the stagnant point on the upper dome and becomes larger on the lower portion of the cylinder and the lower dome. Therefore, because of the large range of Gr_x/Re_x^2 involved, a heat-transfer correlation valid for the entire mixed-convection regime is needed. For simplicity, TMC is assumed to be valid along the entire vessel wall.

An analysis of TMC along a vertical isothermal plate in aiding flow has been reported by Chen et al.[11] The analysis employed a modified Van Driest mixing-length model for turbulent diffusivities that accounts for the buoyancy effect. On the basis of the numerical results of the analysis, they proposed the following correlation for the local Nusselt number [Nu_x (= $h_c x/k$)]:

$$\mathrm{Nu_x\,Re_x^{-0.8}} = F(\mathrm{Pr})\left\{1+0.36\left[\frac{G(\mathrm{Pr})}{F(\mathrm{Pr})}\left(\mathrm{Gr_x}/\mathrm{Re_x^{2.4}}\right)^{1/3}\right]^3\right\}^{1/3} \quad (11)$$

where $\mathrm{Re_x} = u_e x/\nu$

$\mathrm{Gr_x} = g\beta\,\Delta T x^3/\nu^2$

$F(\mathrm{Pr}) = 0.0287\,\mathrm{Pr}^{0.6}$

$G(\mathrm{Pr}) = 0.150\,\mathrm{Pr}^{1/3}[1+(0.492/\mathrm{Pr})^{9/16}]^{-16/27}$

As noted in Ref. 11, Eq. 11 converges to the known result for pure turbulent forced convection over a vertical plate when $Gr_x/Re_x^2 \to 0$; however, it underpredicts the local Nusselt number by about 29% in the pure free convection limit when $Gr_x/Re_x^2 \to \infty$. Chen et al.[11] also noted that, in the free-convection-dominated regime, a better correlation was obtained by replacing the numerical constant 0.36 by 0.52. Pending future analysis of additional PCCS experimental data, the following heat-transfer correlation for TMC is tentatively proposed for this study:

$$\mathrm{Nu_x} = C\,\mathrm{Re_x^{4/5}}\,F(\mathrm{Pr})\left\{1+a\left[\frac{G(\mathrm{Pr})}{F(\mathrm{Pr})}\left(\mathrm{Gr_x}/\mathrm{Re_x^{2.4}}\right)^{1/3}\right]^3\right\}^{1/3} \quad (12)$$

where $C = 1.4$ to account for the fluctuating condensate surface

$$a = \begin{cases} 0.36 \text{ for } \mathrm{Gr_x}/\mathrm{Re_x^2} < 1 \\ 0.52 \text{ for } \mathrm{Gr_x}/\mathrm{Re_x^2} \geq 1 \end{cases}$$

RESULTS AND DISCUSSION

The test conditions and measured performance data for the six tests (numbered 1, 3, 12, 13, 17, and 18) conducted by the Westinghouse Electric Corporation, in which the AP-600 PCCS Small-Scale Test Facility was used, are listed in Ref. 10. Tests 1 and 3 were for dry external walls. The measured data include the average heat flux through the containment vessel wall, the air flow rate through the annulus, the overall evaporation and condensation rates, the vessel pressure, and the temperature and heat-flux distributions along the vessel wall. The overall heat balance was evaluated on the basis of (a) condensate flow rate, (b) evaporating water film and air flow rate in the annulus, and (c) heat-flux meters installed on the vessel wall. The heat-flux measurement was based on the temperature difference across the vessel wall, which was measured with a pair of thermal couples located inside and outside the vessel wall. Large discrepancies among the three methods were reported for tests 1 and 3. The measured data in which heat-flux meters were used were only about one-third of the other [from (a) and (b)] in test 1 and about one-half of the other in test 3 because in these tests the temperature difference across the wall was very small compared with the accuracy of the temperature measurement, so the accuracy of the heat-flux data was very poor. In general, results obtained from methods (a) and (b) showed reasonable agreement for all six tests.

Figure 1 is a sectional view of the Westinghouse AP-600 PCCS Small-Scale Test Facility. Detailed initial and boundary conditions and the numerical model used in the COMMIX calculations are given in Ref. 10.

Transient calculations, in which the TMC model was used, were performed for the conditions corresponding to test 13 of the small-scale tests. Figure 2 shows the increase of the total nondimensional condensation rate (M_c/M_{ss}) on the inside and the total nondimensional evaporation rate (M_e/M_{ss}) on the outside vessel wall surfaces and the corresponding increase of nondimensional pressure (p/p_{ss}) inside the vessel, where M_{ss} and p_{ss} are the steady-state condensation rate and pressure, respectively. Total condensation rate quickly reaches a value close to its steady-state value

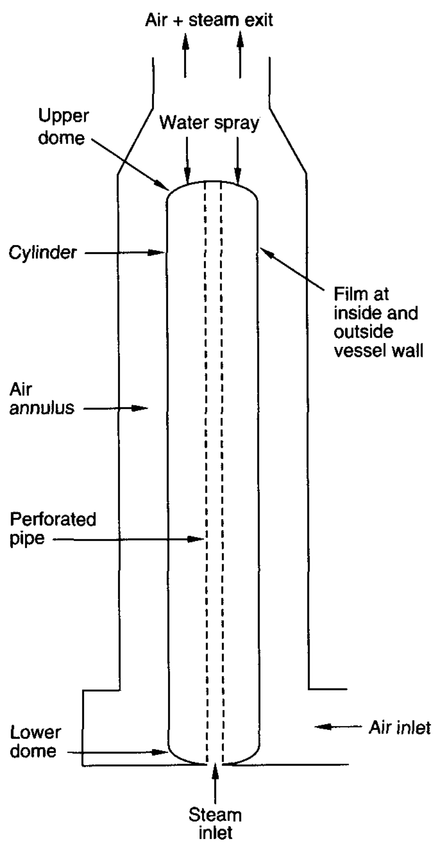


Fig. 1 Vertical section of Westinghouse AP-600 Passive Containment Cooling System Small-Scale Test Facility.

because the vessel wall was initially cold. The evaporation rate increases slowly because of the thermal inertia of the metal vessel wall. Figure 3 shows the steady-state velocity distribution, the nondimensional temperature $(T - T_l)/(T_s - T_l)$, and the steam mass fraction contours in the system, where T_l is the temperature of the water sprayed on the outside of the upper dome and T_s is the temperature of the steam inlet at the bottom of the vessel. As shown in Fig. 3(a), inlet steam flows upward along the centerline in the containment vessel with some spreading. It then flows down along the wall because of both buoyancy and forced convection. In the annulus, the air goes upward with some increase in velocity, mainly because of the addition of steam from the evaporating film. Both temperature and steam concentration in the containment vessel are stratified, as shown in Figs. 3(b) and 3(c).

The time–progressive film thickness distribution on the inside wall of the vessel is plotted in Fig. 4 for t = 5, 10, 20, 30, 50, 70, and 800 s into the transient. Steam condensation starts near the top of the upper dome, which the stream reaches first. The film thickness builds up before it gains enough momentum to flow downward such that a sharp film front can be observed at t = 50 s into the transient. As the film reaches the lower dome, its thickness increases only because of the decreasing surface areas in the flow direction. At steady state (denoted by t = 800 s into the

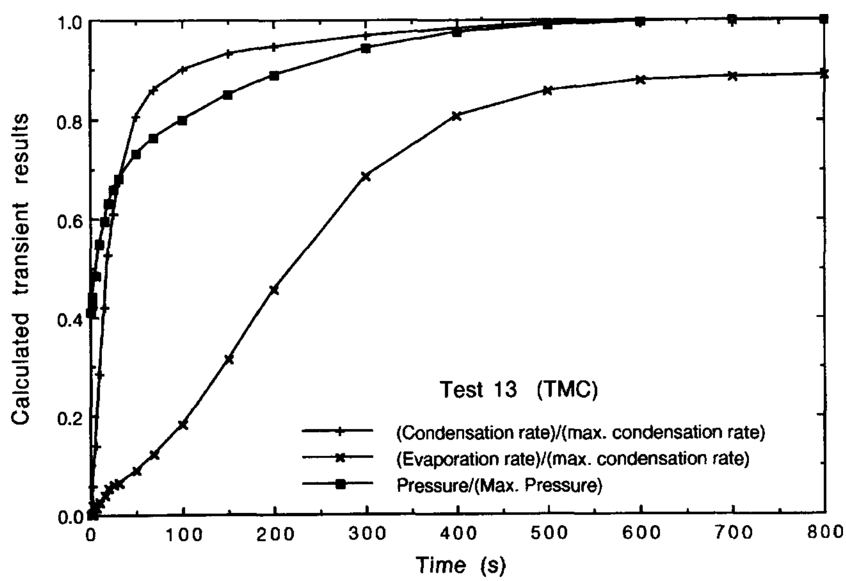
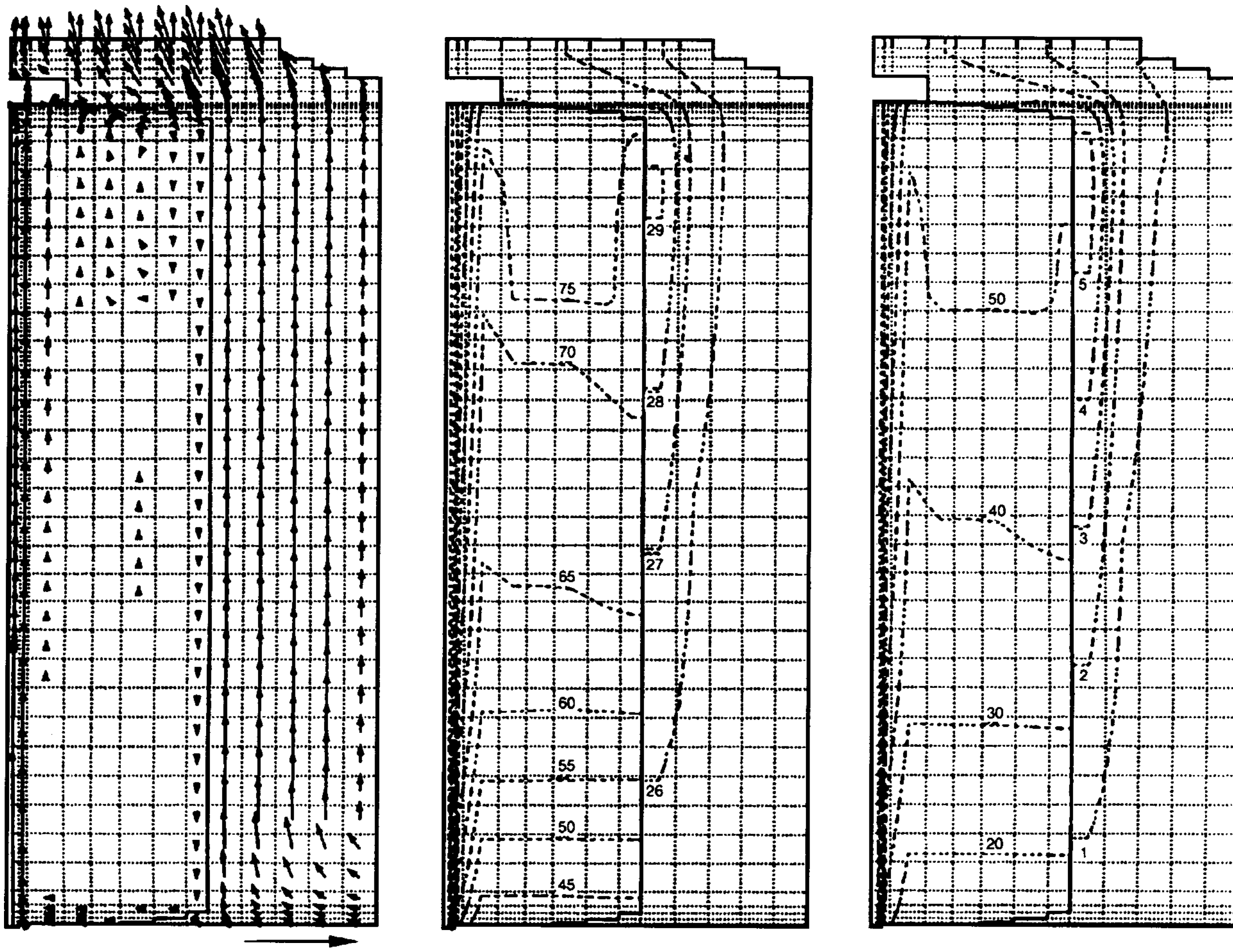


Fig. 2 Illustration showing increase of total condensation and evaporation rates and of vessel pressure with time. Note: TMC is Turbulent-Mixed-Convection model.



Velocity 10.00 m/s
(a)

Nondimensional temperature (%)
(b)

Steam mass fraction (%)
(c)

Fig. 3 COMMIX results of steady-state distributions for Test 13 with Turbulent-Mixed-Convection model.

transient), the film thickness is smaller because higher temperature produces a lower viscosity and hence a higher film-flow velocity. For the evaporating film on the outside surface of the vessel, its thickness distribution is plotted in Fig. 5 for t = 1, 5, 10, 15, 20, 22, and 800 s into the transient. The spray water takes about 22 s to reach the bottom of the cylindrical shell. The steady-state film thickness distribution (denoted by t = 800 s into the transient) is thinner, partially because of the higher film temperature and partially because of water evaporation.

An assessment of the heat-transfer models has been performed for the six tests. In Fig. 6, the calculated vessel pressures, in which the turbulent-free-convection (TFC) correlation (Eq. 8b) and the TMC correlation (Eq. 12) were used, are compared with the measured vessel pressure. Agreement between the measured and calculated pressure with both correlations is quite good for tests 12, 13, and 17. A large discrepancy (16.4%) exists for test 18, however, when the TFC correlation was used, whereas there is only a moderate difference (5.4%) when the TMC correlation was used. The higher predicted pressure for tests 1 and 3 was due to a lack of radiation model in COMMIX on the outside vessel wall. Measured and calculated total evaporation rates are compared in Fig. 7. Here agreement is good between the measured and calculated values for both heat-transfer correlations. In Fig. 8, the calculated

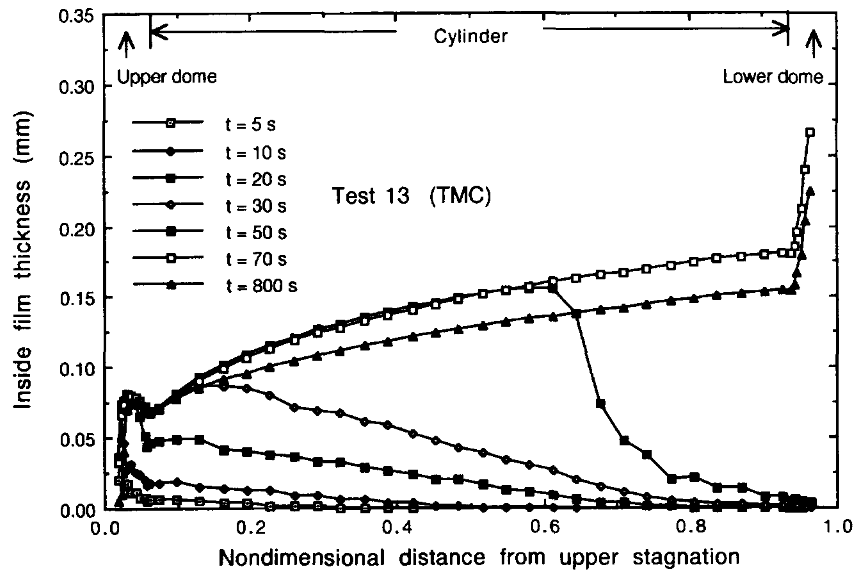


Fig. 4 Illustration showing the thickness of film on inside of vessel wall surface at various times. Note: TMC is Turbulent-Mixed-Convection model.

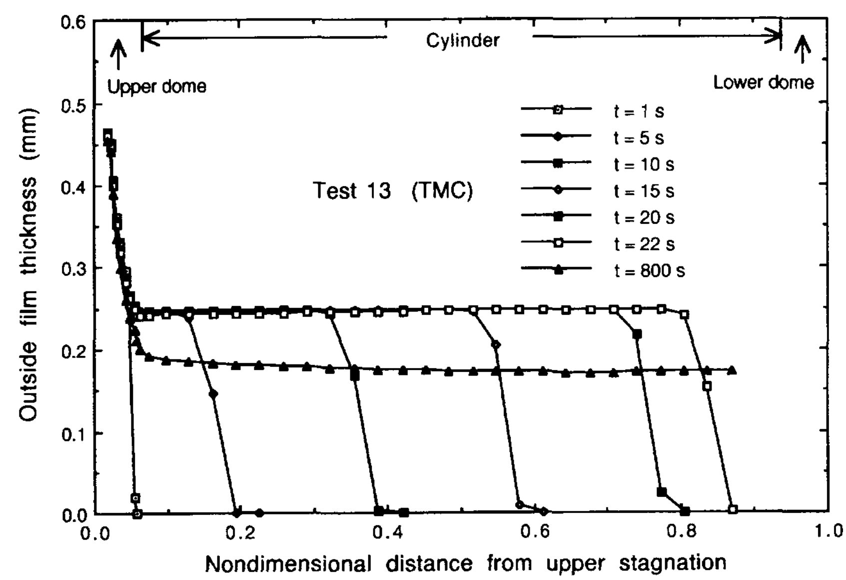


Fig. 5 Illustration showing the thickness of film on outside of vessel wall surface at various times. Note: TMC is Turbulent-Mixed-Convection model.

average heat flux through the vessel wall with the use of the TMC correlation is compared with the measured data evaluated from the three methods described previously. Again, good agreement was obtained.

The measured and calculated streamwise nondimensional wall and film temperature distributions at both the inside and outside surface of the vessel wall for test 18 are shown in Fig. 9. The nondimensional distance ($x^* = x/x_T$) is measured along the vessel surface from the upper stagnation, where x_T is the total distance from upper to lower stagnation. Agreement between measured and calculated data is generally reasonable except

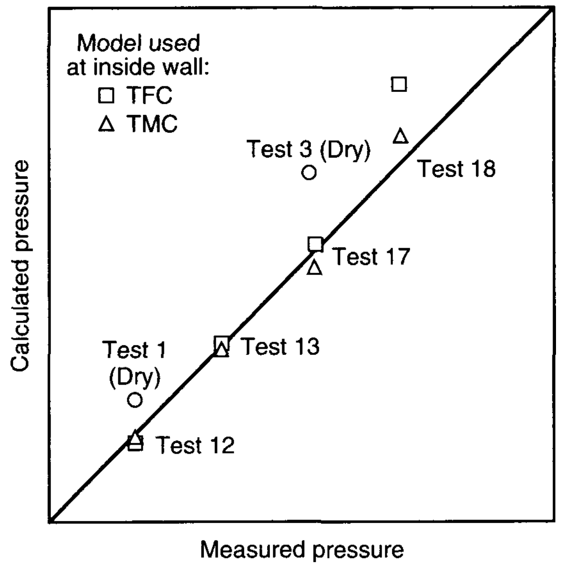


Fig. 6 Comparison of measured and calculated vessel pressure. Note: TMC is Turbulent-Mixed-Convection model and TFC is Turbulent-Free-Convection model.

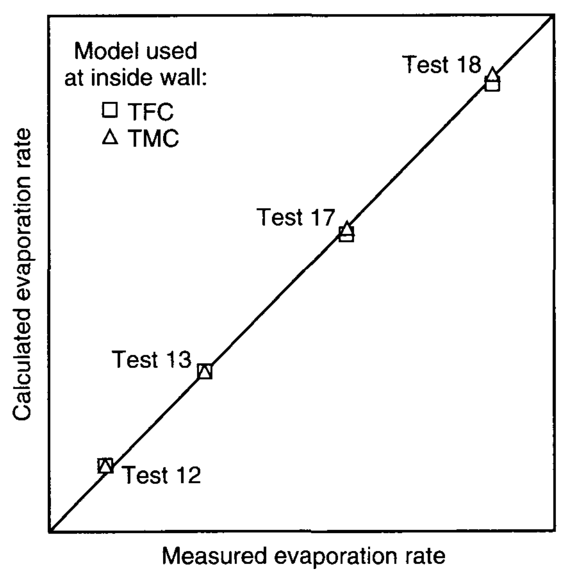


Fig. 7 Comparison of measured and calculated total evaporation. Note: TMC is Turbulent-Mixed-Convection model and TFC is Turbulent-Free-Convection model.

at small x*. In particular, the measured temperatures for $0 < x^* < 0.06$ exhibit a characteristic dip before they increase with x*. The calculated temperature with the use of the TFC correlation [Fig. 9(a)] showed no such phenomena when it was obtained with the TMC correlation [Fig. 9(b)]. The reason is that heat convection is

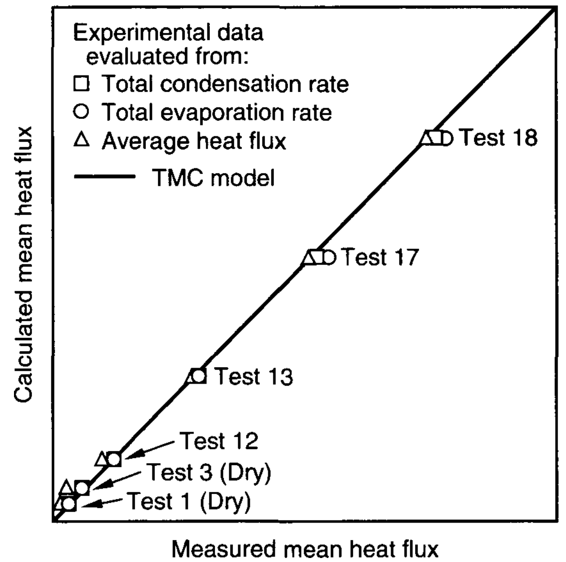


Fig. 8 Comparison of measured and calculated mean heat flux. Note: TMC is Turbulent-Mixed-Convection model.

dominated by forced convection in that region, and the free-convection model underpredicts the heat-transfer coefficient.

The calculated nondimensional heat fluxes ($q^* = q/q_a$) and measured local mean heat fluxes (average of measured fluxes at azimuthal angles 120° apart at each x) on the wall surfaces are plotted in Fig. 10 for test 13. The TMC correlation was used in this calculation. Deviation of the measured data from the mean in the region $0.057 < x^* < 0.15$ is very large. Again, good agreement between measured and calculated results is obtained. Because convective heat flux is usually very small compared with the condensation heat flux, wall heat flux is then approximately proportional to the condensation rate (see Eq. 5c), which further indicates that the heat and mass transfer models used in this study are reasonable. In Fig. 10, the difference of the calculated fluxes on inside and outside is due to the difference in surface areas of the inside and outside vessel walls.

CONCLUSIONS

The COMMIX computer code has been validated by comparing calculated results for the various performance parameters with Westinghouse AP-600 PCCS small-scale test data. To this end, two liquid-film tracking models for time-dependent flows (a simplified model and a comprehensive model) were developed

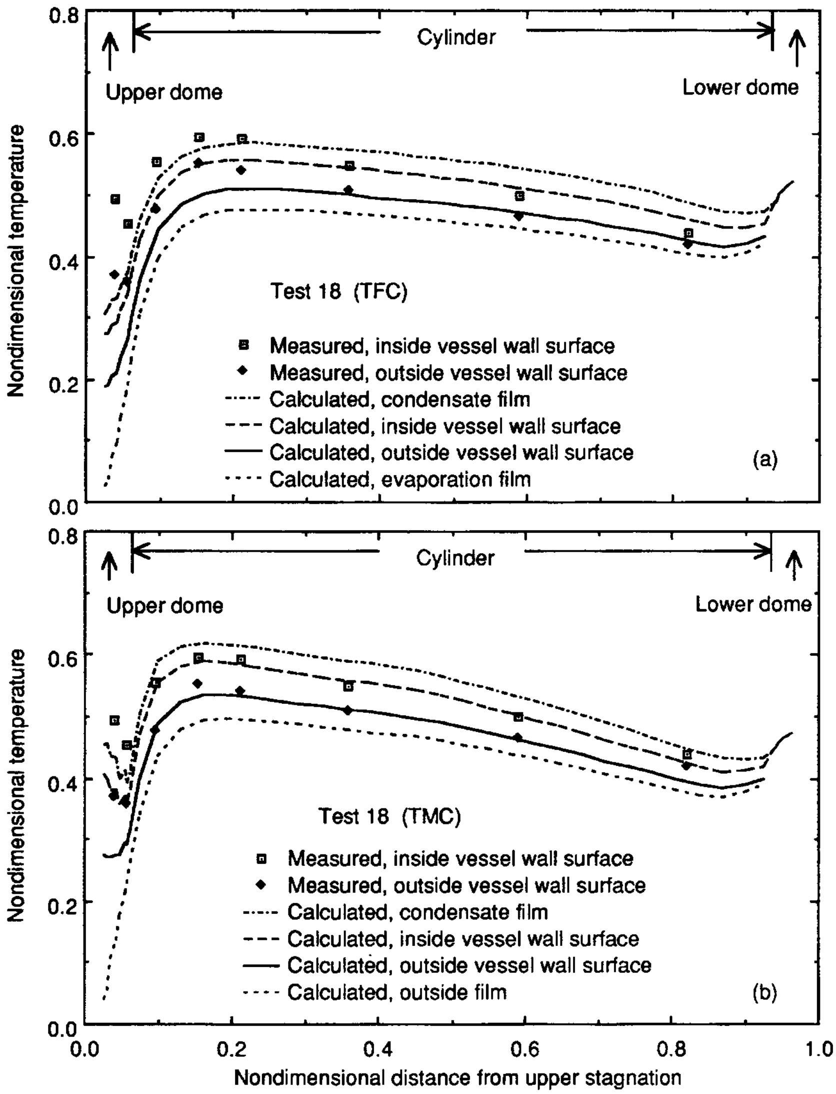


Fig. 9 Temperature on surface of vessel wall and of inside and outside films. Note: TFC is Turbulent-Free-Convection model and TMC is Turbulent-Mixed-Convection model.

and implemented in COMMIX. These models compute transient liquid-film thickness, its mean velocity, and its temperature on both sides of the steel containment vessel. The difference between the simplified and comprehensive models is that the inertia terms in the momentum equation are neglected in the simplified model, whereas they are retained in the comprehensive model. The results obtained by both models were compared and found to be essentially the same.

Six experimental data sets for performance parameters, including average wall heat flux, total evaporation rate, vessel pressure, and streamwise wall temperature and heat-flux distributions, were used in the comparison. The results showed that the steam–air flow inside the containment vessel gave rise to mixed convection. The mixture flow is stratified with more air in the lower portion of the vessel and more steam in the upper portion. Its flow field is complex. With the pure free-convection model and low heat fluxes, reasonable agreement between the experimental and calculated results for the various performance parameters was obtained. At the highest heat flux, calculated vessel pressure differed from the measured vessel pressure by more than 16%. Furthermore, significant differences existed in the

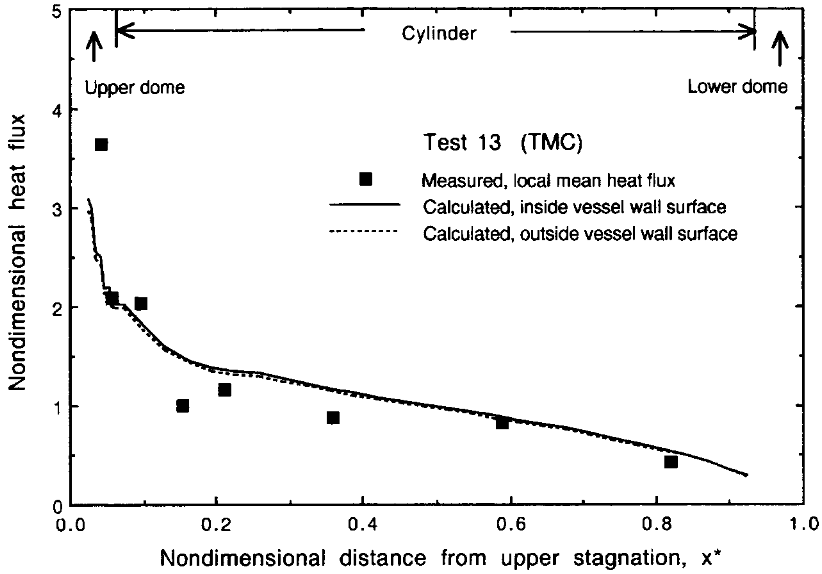


Fig. 10 Comparison of measured and calculated wall heat flux distribution. Note: TMC is Turbulent-Mixed-Convection model.

calculated and measured wall temperature distribution over the upper elliptical dome. An examination of the ratio of the local Grashof number to the square of the local Reynolds number established beyond any doubt for these tests that the steam–air boundary layer flow adjacent to the condensate film was mixed convection. In the present study, a simple TMC model based on the correlation developed by Chen et al.[11] was tentatively proposed. COMMIX assessment was repeated for the conditions of the six Westinghouse experiments, in which the proposed TMC model was used, and compared with the corresponding results from the TFC model used in CONTAIN. The results showed significant improvement in the predicted vessel pressure and wall temperature distribution in the upper dome region at the highest heat flux with the TMC model.

ACKNOWLEDGMENTS

The authors gratefully acknowledge the support and encouragement of Drs. Y. Chen, A. Notafrancesco, A. Rubin, C. Tinkler, F. Altawila, and B. W. Sheron of the Division of System Research, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission. The many stimulating discussions with Professor B. T. Chao of the University of Illinois at Urbana–Champaign were extremely helpful and are much appreciated.

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SYMPOSIUM ON ACCEPTABILITY OF RISK FROM RADIATION—APPLICATION TO MANNED SPACE FLIGHT

Arlington, Virginia, May 29, 1996

The purpose of this symposium is to provide an opportunity for a wide ranging discussion among scientists, radiation biologists, public health professionals, and members of the public on the rationale for establishing an acceptable lifetime risk of fatal cancer due to exposures to ionizing radiation in space. There is no registration fee.

For additional information, contact Laura Atwell, National Council on Radiation Protection and Measurements, Phone (301) 657-2652, or Dade W. Moeller, Phone (919) 633-3352.

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Portland, Oregon, July 15–18, 1996

The 24th DOE/NRC Nuclear Air Cleaning and Treatment Conference will cover all aspects of nuclear airborne waste management and nuclear air and gas cleaning technology. It is expected that a major part of the conference program will consist of contributed papers on (1) new and important developments in nuclear air cleaning research in connection with nuclear power plant operations and waste management and (2) field experiences and applications of special interest to operating personnel in both areas.

Topics of additional interest are: advanced nuclear power plants, decommissioning and demolition of aged nuclear reactors and reprocessing facilities, air cleaning requirements for remediation of sites contaminated with nuclear materials, air and gas cleaning requirements for retrievable and permanent nuclear waste isolation storage facilities, the impact on air cleaning and treatment requirements of recent and pending legislation, regulations, and codes and standards, including ASME AG-1, N509 and N510.

Inquiries regarding the conference should be sent to the Conference Chairman: Melvin W. First, Sc.D., Harvard University Air Cleaning Lab., 665 Huntington Ave., Boston, MA 02115-9957, Tel: (617) 432-1164, Fax: (617) 432-3349, Telex: 501003 HARVINTHLTH

Spent Nuclear Fuel Characterization for a Bounding Reference Assembly for the Receiving Basin for Off-Site Fuel

By S. D. Kahook, R. L. Garrett, L. R. Canas, and M. J. Beckum[a]

Abstract: *The Basis for Interim Operation (BIO) for the Receiving Basin for Off-Site Fuel (RBOF) facility at the Department of Energy (DOE) Savannah River Site (SRS) nuclear materials production complex, developed in accordance with draft DOE-STD-0019-93, required a hazard categorization for the safety analysis section as outlined in DOE-STD-1027-92. The RBOF facility was thus established as a Category-2 facility (having potential for significant on-site consequences from a radiological release) as defined in DOE 5480.23. Given the wide diversity of spent nuclear fuel stored in the RBOF facility, which made a detailed assessment of the total nuclear inventory virtually impossible, the categorization required a conservative calculation based on the concept of a hypothetical, bounding reference fuel assembly integrated over the total capacity of the facility. This scheme not only was simple but also precluded a potential delay in the completion of the BIO.*

The Receiving Basin for Off-Site Fuel (RBOF) at the Department of Energy (DOE) Savannah River Site (SRS) nuclear materials production complex near Aiken, South Carolina, is a facility designed for the receipt, storage, and conditioning of spent nuclear fuel (SNF) from off-site reactors.[1] The facility has been operational since 1963 and is now managed by the Excess Facilities and Reactor Fuel Storage Program Division of the Westinghouse Savannah River Company (WSRC), the prime operations contractor for SRS.

[a]Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina 29808.

OVERVIEW OF THE RECEIVING BASIN FOR OFF-SITE FUEL

The RBOF features a dry cask-wash pit, a cask unloading basin, two fuel storage basins, a fuel inspection basin, a fuel disassembly basin, and a fuel repackaging basin. All the basins are filled with water to variable depths and interconnected by canals. The water in the basins is continuously purified by circulation through a filter–deionizer system.

The main section (22 ft deep) of the large storage basin (27 ft wide by 40 ft long overall) and the small storage basin (13 ft wide by 27 ft long by 29 ft deep) are reserved for intact fuel bundles, which are arranged in rows defined by a vertical framework of racks made of aluminum I-beams rising from the bottom of the basins. Gratings, guide plates, and spacers (collectively referred to as "hardware") are installed between the racks to define the individual storage slots along the rows. Up to four tiers of hardware per row can be installed, depending on the fuel type. The large basin contains 42 rows, each 18 ft long, and the small basin contains 11 rows, each 9 ft long. Twenty-one rows in the large basin and all the rows in the small basin are 9 inches wide; 18 other rows in the large basin are spaced at 12 inches. The remaining 3 rows in the large basin have spacings of 11, 16, and 25.5 inches, respectively.

Fuel assemblies in RBOF are largely bundled in locally fabricated, elongated, aluminum cans (tubes) with either a circular or a squared cross section. The cylindrical cans are commonly referred to as general-purpose (GP) tubes. Assemblies of a specific fuel type are packed in cans as constrained by rigorous nuclear criticality safety criteria. Similar criteria dictate the distribution of up-standing cans throughout the storage basins.

The large basin also has special storage racks known as bucket storage and test tube storage. The first section provides space for 70 buckets in 5 rows; the "buckets" are special containers designed to hold reactor slugs. The second section has a capacity of 13 "test tubes," special containers for damaged (confirmed or suspected) fuel elements. At 29 ft, these sections are somewhat deeper than the main section.

RBOF AUTHORIZATION BASIS UPGRADE

The RBOF Authorization Basis is undergoing extensive revision to conform to current DOE Orders and Standards. The initial task, development of the Basis for Interim Operation (BIO), was originally completed in late 1994 and revised once in early 1995.[2] The BIO is based on DOE-STD-0019-93,[3] although this Standard was later superseded by draft DOE-STD-3011-94.[4] Pending the release of the Safety Analysis Report (SAR), tentatively scheduled for mid-1996, the BIO authorizes operation of the facility within an acceptable safety envelope. The safety analysis portion of the BIO required a hazard categorization as outlined in DOE-STD-1027-92.[5] A chemical–nuclear facility falls into one of three classes defined in DOE 5480.23[6] according to the potential consequences of radiological releases.

THE FUEL DIVERSITY DILEMMA

The RBOF is now loaded to about 85% capacity with a wide variety of SNF, mostly domestic research reactor (DRR) fuel received over the past 30 years. In the foreseeable future, the facility likely will reach full capacity with the increasing influx of foreign research reactor (FRR) fuel from various countries. The SNF inventory (current and expected) is traceable to a wide variety of original designs (chemical–nuclear composition and physical configuration) with an ample diversity of irradiation–cooling history. This variability, coupled with the continuous impact of radioactive decay, complicated the assessment of the nuclear inventory in support of the BIO. A detailed accounting by fuel type and individual assemblies therein required a monumental effort incompatible with the BIO task schedule.

THE RFA CONCEPT AS A PRACTICAL SOLUTION

The aforementioned difficulty with the evaluation of the nuclear inventory in RBOF was circumvented by a conservative calculation based on the concept of a hypothetical, bounding reference fuel assembly (RFA) integrated over the total capacity of the facility. Thus RBOF was conservatively established as a Category-2 facility (having potential for significant on-site consequences from a radiological release). The development of the RFA is herein described as derived from a systematic ranking of the real assemblies according to a maximum burnup criterion. This article focuses on illustrating the technique rather than on providing a detailed quantitative account.

FUEL DESCRIPTION

The RBOF harbors or is expected to receive a wide diversity of DRR and FRR fuel. The facility also contains some production assemblies from SRS reactors. As previously noted, the fuel differs significantly in chemical–nuclear composition, physical configuration, burnup, and cooling. The fuel is stored primarily as bundles of intact assemblies, but some cans contain partial assemblies, assembly elements (such as plates or rods), and fragments.

Table 1 is a representative listing of SNF by element composition and configuration. Fuel in any particular combination of the indicated parameters further differs in nominal (preirradiation) isotopic composition, cladding, element dimensions, and assembly design; Table 2 illustrates such variabilities for the Experimental Boiling-Water Reactor (EBWR) fuel. Moreover, individual assemblies of a given fuel also show significant variations in original composition (relative to a nominal value), burnup, and cooling. Table 3 provides an example based on a shipment of R-2 fuel from Studsvik Nuclear AB (Sweden).

FUEL SCREENING

Summary

The first phase of the work was to rank SNF by the amount of fissile material burned. The results (fuel

Table 1 Partial Listing of Fuel Types in RBOF

Fuel[a]	Composition[b]	Elements[b]
Ames Laboratory Research Reactor	U–Al	Plates
ANL–MXOX (Argonne National Laboratory West)	PuO_2–UO_2	Rods
ASTRA (Austrian Research Centre Seibersdorf)	U–Al	Plates
B&W (Babcock & Wilcox)	U–Al, PuO_2–UO_2	Plates, rods
BMI (Battelle Memorial Institute)	U–Al	Plates
Carolinas Virginia Tube Reactor	UO_2	Rods
Dresden	UO_2, UO_2–ThO_2	Rods
EBR-II	U–Pu	Tubes
EBWR (Experimental Boiling-Water Reactor)	UO_2, U–Zr–Nb	Plates, rods
Elk River	UO_2–ThO_2	Rods
Fermi	U–Mo	Rods
GCRE (Gas-Cooled Reactor Experiment)	UO_2, UO_2–BeO	Pellets
HFIR (High-Flux Isotope Reactor)	U_3O_8–Al	Plates
HTRE (Heat Transfer Experimental Reactor)	UO_2, UO_2–BeO	Capsules
HWCTR (Heavy-Water Components Test Reactor)	Th, U, UO_2, U–Zr	Tubes
Mark-18	Pu–Al, PuO_2–Al	Tubes
Mark-22	U–Al	Tubes
Mark-42	PuO_2–Al	Tubes
MURR (Missouri University Research Reactor)	U–Al	Plates
ORR (Oak Ridge Reactor)	U–Al, U–Al–Si, U_3O_8–Al	Plates
PCA (Pool Critical Assembly)	U–Al, U_3O_8–Al	Plates
R-2 (Studsvik Nuclear, Sweden)	U–Al	Plates
RHF (Reactor á Haut Flux, France)	U–Al	Plates
Saxton	UO_2, PuO_2–UO_2	Rods
SFF/SFO (Sterling Forest Research Center Reactor)	U–Al, U_3O_8–Al	Plates
SPERT-III/SPERT C	UO_2	Pellets
SRE (Sodium Reactor Experiment)	U, U–Mo, UO_2	Slugs
TRR (Taiwan Research Reactor)	U–Al	Tubes
UVA (University of Virginia)	U–Al	Plates
VBWR (Vallecitos Boiling-Water Reactor)	UO_2	Pellets

[a]Some fuels are no longer stored in the RBOF but are cited for completeness.
[b]Most FRR fuel types consist of U–Al plates.

Table 2 Summary of Characteristics of EBWR Fuel

Bundle ID[a]	Total bundles[b]	Bundle configuration	Fuel form	Element dimensions, inches
ET-i	54	1 assembly (6 plates)	U–Zr–Nb 1.44% ^{235}U	0.212 × 3.5/8 × 54
EH-i	50			0.280 × 3.5/8 × 54
EH-17A-II, III	2			
EHS-58	1			
T-i	7		U–Zr–Nb 0.71% ^{235}U	0.212 × 3.5/8 × 54
H-i	11			0.280 × 3.5/8 × 54
ET-11	1	1 assembly (47 plates)	U–Zr–Nb 1.44% ^{235}U	0.280 × 3.5/8 × various lengths
S-i	32	1 assembly (49 rods)	UO_2–ZrO_2–CaO 93.2% ^{235}U	3/8 × 51
E-2-i	59	1 assembly (36 rods)	UO_2 6% ^{235}U	0.430 × 54.21/32
N-i	51		UO_2 0.71% ^{235}U	

[a]Index i stands for sequential numbering.
[b]Inventory as of late 1993.

Table 3 Variations in R-2 Fuel Assemblies[a]

Unit	^{235}U, g Preirradiation	^{235}U, g Postirradiation	Exposure, MWh	Decay heat, W	Discharge date
1	157.82	38.46	2274.8	2.6	12 Jun 1986
2	248.88	111.67	2615.0	3.3	19 Sep 1987
3	157.98	39.45	2259.0	3.2	10 Oct 1987
4	251.01	119.19	2512.3	2.7	14 May 1986
5	247.44	113.88	2545.5	2.8	05 Feb 1987
6	248.27	111.18	2612.7	2.8	28 May 1986
7	250.46	109.74	2681.9	2.7	10 Oct 1986
8	251.29	111.03	2673.1	2.8	12 Aug 1986
9	157.66	32.16	2391.8	3.5	29 Jan 1988
10	250.57	112.18	2637.5	3.0	08 May 1987
11	250.64	112.72	2628.5	3.1	05 Feb 1987
12	250.64	110.55	2669.9	3.0	29 May 1987
13	248.29	107.72	2679.1	3.0	20 Oct 1986
14	251.92	112.02	2666.3	3.1	29 May 1987
15	157.93	33.63	2369.0	4.2	22 Oct 1988
16	157.69	32.93	2377.7	3.8	19 Jun 1988

[a]Representative sample based on one shipment.

names only, values not shown) are shown in Table 4, limited to fuels with the highest *burnup per storage slot* (this parameter is defined in the following section). These fuels are tentatively the worst in terms of the potential consequences from a radiological release.

Table 4 Highest Burnup Fuel Types in RBOF[a]

Fissile material ^{233}U	Fissile material ^{235}U	Fissile material ^{239}Pu
Dresden	Dresden	Mark-42
	MURR	Saxton
	RHF	
	Saxton	

[a]Ranking by fuel type only. Burnup values not shown.

Methodology

The burnup was calculated by subtracting the actinide material content (plutonium, thorium, and uranium—essentially unchanged because the fuel was removed from the source reactors) from the original (preirradiation) amount. The amount of depleted fissile material followed as the difference between the beginning-of-life (BOL) and the end-of-life (EOL) actinide amounts. Ideally, the burnup should have been computed for each individual assembly of each fuel type. In practice, though, this would have proven a monumental task because of the vast diversity of SNF in RBOF. Moreover, the fuel receipt records (especially for the older stock) were either incomplete, missing, or not readily retrievable. In many cases the inventories of interest were available for whole bundles of assemblies only.

Table 5 summarizes the burnup calculations for selected fuel types as extracted from a master spreadsheet. For convenience, SNF was classified as (1) uranium–aluminum alloy (high enrichment), (2) mixed-oxide containing plutonium, (3) uranium-based, (4) plutonium-based, and (5) mixed-oxide containing thorium. In each class the burnup per fuel unit (herein defined as the *specific burnup*) was calculated as shown for a specified inventory of a fuel type. A *fuel unit* stands primarily for a single assembly, but the concept extends to other identifiable fuel entities as constrained by available data or convenience. Differences in fuel units are reconciled by the normalization process described below.

Generally, monthly reports issued by RBOF consolidate the fuel inventory as configured for storage. Such reports provided the basis for ready calculations of *average* specific burnups for different batches of particular fuel types; however, some calculations were extended to

Table 5 Burnup Calculations for Selected DRR Fuel

Category	Fuel	Stock,[a] unit	Material	Initial loading,[b] kg/unit	Stock,[c] kg	Stock,[d] kg/unit	Burnup,[e] kg/unit	Storage capacity,[f] unit/slot	Burnup,[g] kg/slot
U–Al high enrichment	MURR	96 assemblies	^{235}U	7.850×10^{-1}	5.652×10^{1}	5.887×10^{-1}	1.963×10^{-1}	6	1.178
	RHF	4 assemblies		8.654	2.080×10^{1}	5.200	3.454	1	3.454
Mixed oxide (plutonium)	Saxton	1 can	^{239}Pu	6.950×10^{-1}	3.370×10^{-1}	3.370×10^{-1}	3.580×10^{-1}	3	1.074
	EBR-II	30 cans	^{235}U	1.450×10^{-1}	2.550	8.500×10^{-2}	6.000×10^{-2}	5	3.000×10^{-1}
			^{239}Pu	4.870×10^{-2}	1.135	3.783×10^{-2}	1.087×10^{-2}		5.433×10^{-2}
Uranium	Saxton	1 can	^{235}U	2.150	1.071	1.071	1.080	3	3.239
	SPERT-3	2 tubes		6.900×10^{-1}	6.030×10^{-1}	3.015×10^{-1}	3.885×10^{-1}	4	1.554
	SFO	678 cans		2.000×10^{-1}	1.025×10^{2}	1.511×10^{-1}	4.890×10^{-2}	9	4.401×10^{-1}
Plutonium[h]	Mark-i								
Mixed oxide (thorium)	Elk River	189 rods	^{233}U	0.000	1.472×10^{1}	6.680×10^{-3}[i]	-6.680×10^{-3}[j]	128	-8.55×10^{-1}
			^{235}U	4.162×10^{-2}	1.862×10^{2}	2.984×10^{-2}[i]	1.628×10^{-2}		2.084
	Dresden	87.139 rods	^{233}U	0.000		1.183[i]	−1.183[j]	1	−1.183
			^{235}U	3.386		2.255[i]	1.131		1.131

[a]Assemblies, bundles, cans, rods, or tubes.
[b]Nominal loadings from Appendix A in the original fuel receipt agreements.
[c]Data from RBOF nuclear accountability records.
[d]Stock (kg/unit) = stock (kg)/stock (units).
[e]Burnup (kg/unit) = initial loading (kg/unit) – stock (kg/unit).
[f]Storage capacity (units/slot) = GP tube length/unit length. A GP tube is 162.2 inches long. Assembly lengths (not shown) are obtained from Appendix A in the original fuel receipt agreements. The results are rounded down to the nearest integer.
[g]Burnup (kg/slot) = burnup (kg/unit) $*$ storage capacity (units/slot).
[h]These are primarily Mark-i (various designations) assemblies from SRS production reactors. Data for these assemblies have Unclassified Controlled Nuclear Information (UCNI) status or higher and cannot be published. Non-SRS units such as EPR-1 have low ^{239}Pu and ^{235}U burnups bounded by the Mark-i assemblies.
[i]Based on individual data.
[j]^{233}U burnup = initial ^{232}Th converted to ^{233}U minus ^{233}U remaining after irradiation. The reported values are negative because there is no ^{233}U initially.

generate specific burnup distributions on the basis of partitioned inventory data (down to single assemblies in some cases); this approach was appropriate for fuels with substantial burnup variability. The preliminary values derived from input data were next transformed to a storage slot basis via multiplication by the number of fuel units per slot. The different fuel types were then compared on this common (*normalized*) basis.

A *slot* is the minimum amount of physical space required to place a GP tube without regard to the constraints of nuclear criticality safety. On this basis, the overall capacity of RBOF storage basins is about 1700 cans. As was indicated, the number of fuel units in a GP tube depends on the fuel type and is administratively restricted to ensure nuclear criticality safety. For the current purpose, however, the individual capacity was assumed to be the number of fuel units (rounded down to the nearest unit) that could fit lengthwise in a GP tube (162.2 inches) regardless of safety limitations. This approach compensates for dimensional differences among the fuel units.

Exclusions

Certain fuels for which data were insufficient were fully or partially excluded from the master calculation spreadsheet on the premise that they had a low burnup or were unquestionably bounded by other types; for instance, the H. B. Robinson fuel had an initial enrichment of only 0.72% ^{235}U and a total uranium content of only 0.51 kg after exposure. In another case, buckets containing fragments (slugs) of Heavy-Water Components Test Reactor (HWCTR) fuel were bounded by driver assemblies in terms of ^{235}U burned.

Foreign Fuel

The FRR fuel yet to be delivered to RBOF could not be formally screened. To ensure that the RFA (defined in the section on "Reference Fuel Assembly") bounds any FRR fuel, a burnup ceiling of 3000 MWd per storage slot was imposed as an acceptance condition. This value is based on historical data for the highest exposure (about 2600 MWd) plus a 15% margin. On the basis of this criterion and unofficial fuel data, the hypothetical limit was calculated for the number of assemblies per storage slot allowable for particular fuel types. Table 6 summarizes the computations for selected fuels. In each case the required limit far exceeds the physical capacity of a storage slot. Nonetheless, the computational basis will be rigorously verified against the official fuel data as they become available.

Special Cases

Several fuels lack data on residual fissile materials, but their exposure histories are known. In these cases, the ^{235}U burnup was calculated as

Burnup (g) = 1.24 (g/MWd) * Exposure (MWd)

The first factor on the right-hand side is based on 200 MeV per fission:

$$\text{Factor} = \frac{1\,\text{fission}}{200\,\text{MeV}} \times \frac{1\,\text{MeV}}{1.60 \times 10^{-13}\,\text{J}} \times \frac{1 \times 10^{6}\,\text{J}}{\text{MWs}} \times \frac{86\,400\,\text{s}}{\text{day}}$$

$$\times \frac{1\,\text{mol}}{6.023 \times 10^{23}\,\text{atoms}} \times \frac{235\,\text{g}}{\text{mol}} = 1.05\ \text{g/MWd}$$

in turn adjusted for nonfission absorptions by the multiplier 1.169 (ratio of the absorption and fission cross sections for a thermal reaction).

RADIONUCLIDE SPECTRA OF WORST FUEL TYPES

Summary

The second phase encompassed the generation of the radionuclide distribution for the worst fuel types (highest specific burnup) identified in Phase 1. Tables 7 to 9 summarize the results for full cores of the Missouri University Research Reactor (MURR), Reactor á Haut Flux (RHF), and Dresden fuels, respectively. The Saxton fuel is not shown because it is

Table 6 Burnup Calculations for Selected FRR Fuel

Fuel	Stock, assembly	Nominal power, MW	Initial loading,[a] g/assembly	Burnup, %	Burnup,[b] g/assembly	Irradiation, d	Exposure,[c] MWd	Permissible storage,[d] assembly/slot
BER-2 (Germany)	34.5	10	180	0.56	100.8	281.65	2816.50	36
DR-3 (Denmark)	26	10	147	0.50	73.5	154.70	1547.00	50
GRR-1 (Greece)	33.3	5	180	0.30	54.0	291.71	1458.60	68
HIFAR (Australia)	25	10	150	0.39	58.5	118.51	1185.10	63
KUR (Japan)	22.7	5	180	0.24	43.2	159.12	795.60	85
Ljubijana (Slovenia)	83.3	0.25	133	0.15	20.0	5388.60	1347.10	185
MAPLE-X (Canada)	29	10	213	0.55	117.2	275.15	2751.50	31
Orphee (France)	7	14	840	0.30	252.0	102.10	1429.40	14
PARR (Pakistan)	25.4	5	196	0.35	68.6	282.17	1410.80	53
Salazar (Mexico)	100	1	133	0.15	20.0	1616.60	1616.60	185
Seoul-1 (Korea)	12.5	0.25	38	0.15	5.7	230.94	57.73	649

[a] ^{235}U.
[b] Burnup (g/assembly) = initial loading(g/assembly) * burnup (%).
[c] Exposure (MWd) = irradiation (d) * nominal power (MW).
[d] Permissible storage (assembly/slot) = stock * 3000/exposure. Results are rounded down to nearest whole assembly.

Table 7 Actinide and Fission-Product Activities for MURR Core[a,b]

Nuclide	Type of irradiation: Uniform[c]	Type of irradiation: Cycled[d]	Nuclide	Type of irradiation: Uniform	Type of irradiation: Cycled
^{3}H	1.66×10^{1}	1.63×10^{1}	^{144}Ce	2.13×10^{4}	1.76×10^{4}
^{85}Kr	4.61×10^{2}	4.54×10^{2}	^{144}Pr	2.13×10^{4}	1.76×10^{4}
^{89}Sr	1.57×10^{1}	8.06	^{144m}Pr	2.56×10^{2}	2.11×10^{2}
^{90}Sr	3.91×10^{3}	3.89×10^{3}	^{147}Pm	8.52×10^{3}	7.86×10^{3}
^{90}Y	3.92×10^{3}	3.89×10^{3}	^{148m}Pm	1.19×10^{-2}	1.13×10^{-2}
^{91}Y	7.15×10^{1}	3.83×10^{1}	^{151}Sm	2.49×10^{1}	2.95×10^{1}
^{95}Zr	1.57×10^{2}	8.64×10^{1}	^{154}Eu	5.20×10^{1}	5.28×10^{1}
^{95}Nb	3.48×10^{2}	1.92×10^{2}	^{155}Eu	6.18×10^{1}	6.08×10^{1}
^{95m}Nb	1.16	6.41×10^{-1}	^{231}Th	9.90×10^{-3}	9.90×10^{-3}
^{99}Tc	5.90×10^{-1}	5.95×10^{-1}	^{234}Th	1.54×10^{-4}	1.54×10^{-4}
^{103}Ru	6.42×10^{-1}	3.11×10^{-1}	^{233}Pa	4.81×10^{-3}	4.82×10^{-3}
^{103m}Rh	5.78×10^{-1}	2.80×10^{-1}	^{234m}Pa	1.54×10^{-4}	1.54×10^{-4}
^{106}Ru	1.97×10^{3}	1.70×10^{3}	^{234}U	4.28×10^{-4}	4.35×10^{-4}
^{106}Rh	1.97×10^{3}	1.70×10^{3}	^{235}U	9.90×10^{-3}	9.90×10^{-3}
^{110m}Ag	8.25×10^{-1}	7.21×10^{-1}	^{236}U	1.96×10^{-2}	1.96×10^{-2}
^{119m}Sn	8.66×10^{-1}	6.98×10^{-1}	^{237}U	6.52×10^{-5}	Nil
^{123}Sn	4.93	3.39	^{238}U	1.54×10^{-4}	1.54×10^{-4}
^{125}Sb	1.76×10^{2}	1.66×10^{2}	^{237}Np	4.81×10^{-3}	4.82×10^{-3}
^{125m}Te	4.29×10^{1}	4.05×10^{1}	^{236}Pu	8.32×10^{-4}	9.15×10^{-4}
^{127}Te	1.06×10^{1}	6.97	^{238}Pu	5.29	6.12
^{127m}Te	1.09×10^{1}	7.11	^{239}Pu	9.75×10^{-2}	9.68×10^{-2}
^{129}Te	1.60×10^{-3}	7.53×10^{-4}	^{240}Pu	4.58×10^{-2}	4.66×10^{-2}
^{129m}Te	2.46×10^{-3}	1.16×10^{-3}	^{241}Pu	2.66	2.71
^{134}Cs	9.41×10^{2}	9.49×10^{2}	^{241}Am	9.28×10^{-3}	9.93×10^{-3}
^{137}Cs	4.07×10^{3}	4.05×10^{3}	^{242}Cm	1.09×10^{-3}	2.49×10^{-3}
^{137m}Ba	3.85×10^{3}	3.83×10^{3}	^{244}Cm	1.74×10^{-4}	1.80×10^{-4}
^{141}Ce	8.61×10^{-2}	4.04×10^{-2}			

[a]MURR core is made up of 8 assemblies.

[b]Computations using ORIGEN 2.1 code with pwrus cross-section library.

[c]Uniform exposure of 120 d at 11 MW (1320 MWd). All figures are in curies (Ci) for a cooling period of 2 years. Total activity = 7.35×10^{4} Ci. Omitted values are <0.001% of the overall activity.

[d]Twenty-four cycles of 5 d at 11 MW and 7.5 d at zero power. All figures are in curies (Ci) for a cooling period of 2 years. Total activity = 6.45×10^{4} Ci. Values labeled negligible are <0.001% of the overall activity.

bounded by the other types. The Mark-42 results have Unclassified Controlled Nuclear Information (UCNI) status and are therefore purposely omitted.

Preliminary Analysis

Because of the lack of exact data (power, irradiation length, and neutron spectrum) needed to simulate properly the burnup and depletion of fuel assemblies, a parametric study was first effected to compare the activities of actinide and fission products at 2 years following the irradiation of 10-kg masses of ^{233}U, ^{235}U, and ^{239}Pu. This cooling period is a lower bound for all fuels currently stored in RBOF and expected in the foreseeable future; DOE now requires RBOF to ensure the robustness of the BIO and, eventually, the SAR. For each material, the study evaluated the resulting activities for various combinations of power, irradiation length, and cross sections at a constant exposure of 3000 MWd. In turn, the results were weighted per assumed release fractions (0.1% for actinides/daughters and 100% for fission products) and inhalation committed dose-equivalent values.[7] Computations were performed with the Oak Ridge Isotope Generation and Depletion Code (ORIGEN).[8]

Table 8 Actinide and Fission-Product Activities for RHF Core[a,b]

	D_2O neutron spectrum[c]			D_2O neutron spectrum	
Nuclide	Natural U	Enriched U	Nuclide	Natural U	Enriched U
^{3}H	3.41×10^{1}	3.41×10^{1}	^{144}Pr	4.78×10^{4}	4.78×10^{4}
^{85}Kr	9.48×10^{2}	9.49×10^{2}	^{144m}Pr	5.73×10^{2}	5.74×10^{2}
^{89}Sr	4.92×10^{1}	4.92×10^{1}	^{147}Pm	1.88×10^{4}	1.87×10^{4}
^{90}Sr	8.08×10^{3}	8.06×10^{3}	^{151}Sm	1.97×10^{1}	1.99×10^{1}
^{90}Y	8.08×10^{3}	8.07×10^{3}	^{154}Eu	1.58×10^{2}	1.61×10^{2}
^{91}Y	2.13×10^{2}	2.13×10^{2}	^{155}Eu	9.34×10^{1}	9.47×10^{1}
^{95}Zr	4.54×10^{2}	4.54×10^{2}	^{231}Th	1.14×10^{-2}	1.14×10^{-2}
^{95}Nb	1.01×10^{3}	1.01×10^{3}	^{234}Th	2.16×10^{-4}	2.16×10^{-4}
^{95m}Nb	3.37	3.37	^{233}Pa	2.71×10^{-3}	2.76×10^{-3}
^{103}Ru	2.17	2.17	^{234m}Pa	2.16×10^{-4}	2.16×10^{-4}
^{103m}Rh	1.96	1.96	^{234}U	1.50×10^{-4}	1.58×10^{-4}
^{106}Ru	4.13×10^{3}	4.15×10^{3}	^{235}U	1.14×10^{-2}	1.14×10^{-2}
^{106}Rh	4.13×10^{3}	4.15×10^{3}	^{236}U	3.28×10^{-2}	3.29×10^{-2}
^{119m}Sn	1.42	1.46	^{237}U	4.40×10^{-5}	4.17×10^{-5}
^{123}Sn	1.01×10^{1}	1.02×10^{1}	^{238}U	2.16×10^{-4}	2.16×10^{-4}
^{125}Sb	3.03×10^{2}	3.07×10^{2}	^{237}Np	2.71×10^{-3}	2.76×10^{-3}
^{125m}Te	7.39×10^{1}	7.49×10^{1}	^{236}Pu	3.63×10^{-5}	4.00×10^{-5}
^{127}Te	2.38×10^{1}	2.40×10^{1}	^{238}Pu	1.56	1.62
^{127m}Te	2.43×10^{1}	2.45×10^{1}	^{239}Pu	8.53×10^{-2}	8.72×10^{-2}
^{134}Cs	1.43×10^{3}	1.48×10^{3}	^{240}Pu	4.32×10^{-2}	4.61×10^{-2}
^{137}Cs	8.34×10^{3}	8.34×10^{3}	^{241}Pu	1.79	1.70
^{134}Cs	9.41×10^{2}	7.89×10^{3}	^{241}Am	6.10×10^{-3}	5.79×10^{-3}
^{137m}Ba	7.89×10^{3}	4.78×10^{4}	^{242}Cm	3.22×10^{-4}	3.14×10^{-4}
^{144}Ce	4.78×10^{4}	4.78×10^{4}	^{244}Cm	4.32×10^{-5}	4.68×10^{-5}

[a]RHF core consists of 2 fuel tubes with a total of 280 plates.

[b]Computations using ORIGEN 2.1 code with candunau (natural uranium) and canduseu (enriched uranium) cross-section libraries.

[c]Uniform exposure of 50 d at 62.7 MW (3135 MWd) for both cross-section libraries. All figures are in curies (Ci) for a cooling period of 2 years. Total activity = 1.60×10^{5} Ci for either natural or enriched uranium.

The study provided the following insights:

- The activities of specific actinides and fission products are highly sensitive (>50%) to the neutron spectra during irradiation, but the spreads are largely damped (<3%) upon weighting the results per radiological hazards.
- For ^{233}U and ^{235}U, the actinide levels increase and the fission-product levels decrease with increasing irradiation length (and a corresponding power decrease per the imposed constant exposure). For ^{239}Pu, the activities of both groups decrease. In all cases, however, the overall radiological hazards decrease with increasing irradiation period.
- Plutonium-239 has a higher potential radiological hazard than ^{233}U and ^{235}U. The difference narrows with increasing irradiation length.

Methodology

Depletion–burnup calculations were effected with the ORIGEN 2.1 code;[8] the specific cross-section libraries are cited in the tabulated results. This code employs the matrix exponential method to compute the buildup, decay, and processing of radioactive materials. In each case the code accepts input data in the form of BOL composition, EOL composition, irradiation history (power level and irradiation length), and reactor-specific parameters (maximum power level, number of fuel assemblies, etc.). These variables are not all independent; the analyst prescribes the appropriate set according to the information on hand. The code also provides multiple options for calculation management and output.

It is not practical to elaborate on the specifics of each fuel type here; therefore this section is limited

Table 9 Actinide and Fission-Product Activities for Dresden Core[a,b]

Nuclide	Activity[c]	Nuclide	Activity	Nuclide	Activity
^{3}H	4.39×10^{1}	^{147}Pm	1.52×10^{4}	^{229}Th	8.74×10^{-3}
^{85}Kr	1.05×10^{3}	^{151}Sm	6.94×10^{1}	^{232}Th	1.72×10^{-2}
^{89}Sr	1.54×10^{1}	^{154}Eu	2.51×10^{2}	^{231}Pa	2.28×10^{-1}
^{90}Sr	7.68×10^{3}	^{155}Eu	2.66×10^{2}	^{233}Pa	8.59×10^{-2}
^{90}Y	7.68×10^{3}	^{208}Tl	8.46	^{232}U	4.09×10^{1}
^{91}Y	7.05×10^{1}	^{209}Pb	8.47×10^{-3}	^{233}U	3.93×10^{1}
^{95}Zr	1.74×10^{2}	^{211}Pb	1.66×10^{-2}	^{234}U	1.92
^{95}Nb	3.86×10^{2}	^{212}Pb	2.36×10^{1}	^{235}U	8.87×10^{-3}
^{99}Tc	Nil	^{211}Bi	1.66×10^{-2}	^{237}U	2.59×10^{-1}
^{106}Ru	1.32×10^{4}	^{212}Bi	2.36×10^{1}	^{238}U	8.42×10^{-2}
^{106}Rh	1.32×10^{4}	^{213}B	8.74×10^{-3}	^{237}Np	8.81×10^{-3}
^{110}Ag	Nil	^{212}Po	1.51×10^{1}	^{239}Np	1.94×10^{-1}
^{110m}Ag	1.88×10^{1}	^{213}Po	8.55×10^{-3}	^{236}Pu	Nil
^{113m}Cd	3.35	^{215}Po	1.66×10^{-2}	^{238}Pu	5.19×10^{1}
^{119m}Sn	3.93	^{216}Po	2.36×10^{1}	^{239}Pu	5.80×10^{1}
^{123}Sn	1.45×10^{1}	^{217}At	8.74×10^{-3}	^{240}Pu	4.60×10^{1}
^{125}Sb	8.70×10^{2}	^{219}Rn	1.66×10^{-2}	^{241}Pu	1.06×10^{4}
^{125m}Te	2.12×10^{2}	^{220}Rn	2.36×10^{1}	^{241}Am	3.97×10^{1}
^{127}Te	3.47×10^{1}	^{221}Fr	8.74×10^{-3}	^{242m}Am	3.41×10^{-1}
^{127m}Te	3.54×10^{1}	^{223}Ra	1.66×10^{-2}	^{242}Am	3.40×10^{-1}
^{134}Cs	3.50×10^{3}	^{224}Ra	2.36×10^{1}	^{243}Am	1.94×10^{-1}
^{137}Cs	9.28×10^{3}	^{225}Ra	8.74×10^{-3}	^{242}Cm	2.69×10^{1}
^{137m}Ba	8.78×10^{3}	^{225}Ac	8.74×10^{-3}	^{243}Cm	Nil
^{144}Ce	3.63×10^{4}	^{227}Ac	1.71×10^{-2}	^{244}Cm	6.74
^{144}Pr	3.63×10^{4}	^{227}Th	1.64×10^{-2}	^{246}Cm	Nil
^{144m}Pr	4.35×10^{2}	^{228}Th	2.35×10^{1}		

[a]Hypothetical, composite core based on the highest activities for the individual nuclides out of computational runs for three actual Dresden containers stored in RBOF.

[b]Computations using ORIGEN 2.1 code with bwrus cross-section library.

[c]All figures are in curies (Ci). Total activity = 1.66×10^{5} Ci. Values labeled as negligible are <0.001% of the overall activity.

to the MURR fuel for illustration. MURR is a pressurized-water vessel contained in an open pool. Light water acts as both moderator and coolant. The reactor has a core of eight assemblies, each of which consists of 24 curved, U–Al plates.[9] Table 10 lists the BOL assembly and core compositions. The assemblies are recycled in and out of the core about 24 times during their lifetime; each cycle consists of 5 days at power and 7.5 days cooling for an overall cycle of 300 days. The reactor operates at a normal power level of 10 MW with a specific power of 1.613 kW/kg ^{235}U. Table 11 shows the EOL compositions and burnups of the highest exposed assemblies out of seven fuel shipments to RBOF, each shipment of which consisted of eight assemblies (full

Table 10 BOL MURR Fuel Composition[a]

Material	Assembly		Core
	Nominal	Maximum	maximum
^{235}U	775.0	782.8	6 262.0
U	832.0	840.3	6 723.0
^{238}U[b]	57.0	57.6	461.0
^{239}Pu	0.0	0.0	0.0
Th	0.0	0.0	0.0
Al (fuel)	866.2	866.2	6 930.0
Al (clad)	3 012.1	3 012.1	24 097.0

[a]All figures are in grams (g).

[b]Assumed as balance upon subtraction of ^{235}U.

Table 11 EOL MURR Fuel Composition

	Shipment						
Material[a]	**1**	**2**	**3**	**4**	**5**	**6**	**7**
^{237}Np	5.42	5.420	5.420	5.420	5.420	5.420	5.420
^{235}U	582.82	584.470	587.260	584.100	584.180	585.590	581.700
^{236}U	29.99	29.860	29.440	29.970	29.970	29.610	29.960
^{238}U	56.70	56.690	56.710	56.710	56.710	56.710	56.630
Pu	0.20	0.200	0.200	0.200	0.200	0.200	0.200
Exposure[b]	149.98	149.760	147.240	147.450	149.890	148.040	149.800
Irradiation[c]	120.00	120.000	120.000	120.000	120.000	120.000	120.000
Power[d]	1.25	1.248	1.227	1.229	1.249	1.234	1.248

[a]All figures are in grams (g).
[b]Exposure (MWd) = irradiation (d) × power (MW).
[c]Irradiation length in days.
[d]Power in megawatts (MW).

Table 12 Radionuclide Distribution of RBOF Reference Fuel Assembly

Nuclide	Activity[a]	Nuclide	Activity	Nuclide	Activity
^{3}H	5.16×10^{1}	^{144}Pr	4.78×10^{4}	^{231}Th	1.14×10^{-2}
^{85}Kr	1.05×10^{3}	^{144m}Pr	5.74×10^{2}	^{232}Th	1.72×10^{-2}
^{89}Sr	4.92×10^{1}	^{147}Pm	1.88×10^{4}	^{234}Th	2.16×10^{-4}
^{90}Sr	8.08×10^{3}	^{148m}Pm	8.93×10^{-3}	^{231}Pa	2.28×10^{-1}
^{90}Y	8.08×10^{3}	^{151}Sm	6.94×10^{1}	^{233}Pa	8.59×10^{-2}
^{91}Y	2.13×10^{2}	^{154}Eu	7.27×10^{2}	^{234m}Pa	2.16×10^{-4}
^{95}Zr	4.54×10^{2}	^{155}Eu	3.81×10^{2}	^{232}U	4.09×10^{1}
^{95}Nb	1.01×10^{3}	^{208}Tl	8.46	^{233}U	3.93×10^{1}
^{95m}Nb	3.37	^{209}Pb	8.74×10^{-3}	^{234}U	1.92
^{99}Tc	1.03	^{211}Pb	1.66×10^{-2}	^{235}U	1.14×10^{-2}
^{103}Ru	2.17	^{212}Pb	2.36×10^{1}	^{236}U	3.29×10^{-2}
^{103m}Rh	1.96	^{211}Bi	1.66×10^{-2}	^{237}U	2.59×10^{-1}
^{106}Ru	2.11×10^{4}	^{212}Bi	2.36×10^{1}	^{238}U	8.42×10^{-2}
^{106}Rh	2.11×10^{4}	^{213}B	8.74×10^{-3}	^{237}Np	8.81×10^{-3}
^{110}Ag	2.32	^{212}Po	1.51×10^{1}	^{239}Np	9.62
^{110m}Ag	1.74×10^{2}	^{213}Po	8.55×10^{-3}	^{236}Pu	1.12×10^{2}
^{113m}Cd	6.955	^{215}Po	1.66×10^{-2}	^{238}Pu	5.19×10^{1}
^{119m}Sn	3.93	^{216}Po	2.36×10^{1}	^{239}Pu	5.80×10^{1}
^{123}Sn	1.45×10^{1}	^{217}At	8.74×10^{-3}	^{240}Pu	9.78×10^{3}
^{125}Sb	8.70×10^{2}	^{219}Rn	1.66×10^{-2}	^{241}Pu	1.06×10^{4}
^{125m}Te	2.12×10^{2}	^{220}Rn	2.36×10^{1}	^{241}Am	5.17×10^{1}
^{127}Te	3.47×10^{1}	^{221}Fr	8.74×10^{-3}	^{242m}Am	3.41×10^{-1}
^{127m}Te	3.54×10^{1}	^{223}Ra	1.66×10^{-2}	^{242}Am	3.40×10^{-1}
^{129}Te	1.20×10^{-3}	^{224}Ra	2.36×10^{1}	^{243}Am	9.62
^{129m}Te	1.85×10^{-3}	^{225}Ra	8.74×10^{-3}	^{242}Cm	4.90×10^{2}
^{134}Cs	1.03×10^{4}	^{225}Ac	8.74×10^{-3}	^{243}Cm	4.90
^{137}Cs	9.28×10^{3}	^{227}Ac	1.71×10^{-2}	^{244}Cm	2.75×10^{3}
^{137m}Ba	8.78×10^{3}	^{227}Th	1.64×10^{-2}	^{246}Cm	2.15×10^{-1}
^{141}Ce	6.46×10^{-2}	^{228}Th	2.35×10^{1}		
^{144}Ce	4.78×10^{4}	^{229}Th	8.74×10^{-3}		

[a]All figures are in curies (Ci). Total activity = 2.31×10^{5} Ci.

cores). The maximum exposed assembly among the ones compared was used as the basis for calculations.

With the preceding information on hand, the radionuclide distribution for the MURR core (Table 7) was generated for the following scenarios: (1) normal cycled operation as described earlier but at a power level of 11 MW (10% increase over nominal) and (2) uniform irradiation at 11 MW for 120 days (same overall exposure of 1320 MWd).

REFERENCE FUEL ASSEMBLY

Summary

With the worst (highest specific burnup) fuels and their corresponding radionuclide distribution established, the last phase was the definition of the desired bounding RFA. Table 12 shows the radionuclide distribution for the RFA.

Methodology

The radionuclide distribution for the RFA was constructed from the highest activities of the individual radionuclides in the distributions of the worst fuels (MURR, RHF, Dresden, and Mark-42) as shown in Tables 7 to 9. (The Mark-42 fuel is properly accounted for but not explicitly shown because of its UCNI status.) The MURR distribution (based on a full core of eight assemblies) was first adjusted by the factor 6/8 throughout to compensate for the actual capacity (six assemblies) of a storage slot in RBOF.

CONCLUSIONS

As derived from the highest burnup fuels at minimum cooling, the RFA unquestionably bounds any single assembly, or bundle of assemblies in a single storage slot, of any fuel now stored in RBOF or expected in the foreseeable future. Moreover, the nuclear inventory arising from one RFA integrated over the hypothetical, conservative capacity of RBOF (1700 slots) is absolutely bounding. (An even more conservative capacity of 2200 slots was assumed to establish RBOF as a Category-2 facility.) In the context of the applicability of RFA, the safety documentation is extremely robust and likely will remain valid for the rest of the useful life of RBOF.

The RFA concept illustrates a simple, elegant, and cost-effective solution to a uniquely complex situation. As such, it is potentially applicable to analogous scenarios in both the nuclear and the nonnuclear sectors. In the course of the review and approval of the RBOF BIO, DOE not only endorsed the RFA notion but also imposed a minimum cooling requirement of 2 years for fuel received in RBOF.

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Operating Experiences

Edited by G. A. Murphy

Reactor Shutdown Experience

Compiled by J. W. Cletcher[a]

This section presents a regular report of summary statistics relating to recent reactor shutdown experience. The information includes both numbers of events and rates of occurrence. It was compiled from data about operating events entered into the SCSS data system by the Nuclear Operations Analysis Center at the Oak Ridge National Laboratory and covers the six-month period of January 1 to June 30, 1995. 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 (Jan. 1, 1995) are included. The second column for each reactor type shows the annualized shutdown rate for the reporting period. The third and fourth columns list cumulative data (numbers and rates) starting as of May 1, 1984.

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

	BWRs (37)				PWRs (76)			
Reactor power (P), %	Number	Shutdown rate (annualized for period)	Cumulative number	Cumulative shutdown rate per reactor year[b]	Number	Shutdown rate (annualized for period)	Cumulative number	Cumulative shutdown rate per reactor year[c]
0	8	0.44	699	1.77	5	0.13	475	0.61
$0 < P \leq 10$	1	0.05	138	0.35	3	0.08	173	0.22
$10 < P \leq 40$	1	0.05	165	0.42	2	0.05	323	0.41
$40 < P \leq 70$	2	0.11	157	0.40	0	0.00	178	0.23
$70 < P \leq 99$	8	0.44	385	0.98	6	0.16	518	0.66
$99 < P \leq 100$	20	1.09	500	1.27	32	0.86	1206	1.54
Total	40	2.18	2044	5.18	48	1.29	2873	3.66

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

[b]Based on cumulative BWR operating experience of 394.76 reactor years.

[c]Based on cumulative PWR operating experience of 784.57 reactor years.

[a]Oak Ridge National Laboratory.

Table 2 shows data on shutdowns by shutdown type: *Shutdowns required by Technical Specifications* are automatic scrams under circumstances where such a shutdown was required; *Intentional or required manual reactor protection system actuations* are manual shutdowns in which the operators, for reasons that appeared valid to them, took manual actions to actuate features of the reactor protection system; *Required automatic reactor protection system actuations* are actuations that the human operators did not initiate but 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 (Jan. 1, 1995, for this issue). The first line of this table gives the information for reactors licensed for full power but not yet in commercial operation on that date.

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

Shutdown (SD) type	BWRs (37) Number	BWRs (37) Shutdown rate (annualized for period)	BWRs (37) Cumulative number	BWRs (37) Cumulative shutdown rate per reactor year[b]	PWRs (76) Number	PWRs (76) Shutdown rate (annualized for period)	PWRs (76) Cumulative number	PWRs (76) Cumulative shutdown rate per reactor year[c]
SDs required by Technical Specifications	7	0.38	268	0.68	3	0.08	414	0.53
Intentional or required manual reactor protection system actuations	11	0.60	207	0.52	18	0.48	389	0.50
Required automatic reactor protection system actuations	18	0.98	947	2.40	25	0.67	1618	2.06
Unintentional or unrequired manual reactor protection system actuations	0	0.00	9	0.02	1	0.03	20	0.03
Unintentional or unrequired automatic reactor protection system actuations	4	0.22	613	1.55	1	0.03	432	0.55
Total	40	2.18	2044	5.18	48	1.29	2873	3.66

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

[b]Based on cumulative BWR operating experience of 394.76 reactor years.

[c]Based on cumulative PWR operating experience of 784.57 reactor years.

Table 3 Reactor Shutdowns by Reactor Type and Reactor Age[a]
(Period Covered is the Second Half of 1994)

Years in commercial operation (C.O.)	BWRs (37)						PWRs (76)					
	Exposure during the period (in reactor years)	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]	0.500	1	0	0.00	330	21.32	0.000	0	0	0.00	336	34.24
First year of C.O.	0.000	0	0	0.00	121	9.00	0.000	0	0	0.00	281	9.96
Second through fourth year of C.O.	0.000	0	0	0.00	264	6.29	0.500	1	1	2.02	529	5.55
Fifth through seventh year of C.O.	0.740	3	4	5.37	185	4.25	3.490	8	7	2.01	332	3.17
Eighth through tenth year of C.O.	3.830	9	9	2.35	222	4.92	7.260	15	5	0.69	391	3.47
Eleventh through thirteenth year of C.O.	2.360	5	9	3.81	282	5.54	3.540	9	4	1.13	506	4.08
Fourteenth through sixteenth year of C.O.	0.500	1	3	6.05	400	6.16	3.060	7	5	1.64	375	3.19
Seventeenth through nineteenth year of C.O.	0.990	2	1	1.01	282	4.47	5.340	11	7	1.31	272	2.56
Twentieth through twenty-second year of C.O.	4.780	11	8	1.67	175	4.02	8.870	19	11	1.24	120	1.89
Twenty-third through twenty-fifth year of C.O.	3.590	8	3	0.84	63	3.18	4.130	9	8	1.94	41	2.02
Twenty-sixth through twenty-eighth year of C.O.	1.050	3	1	0.95	9	2.13	0.990	2	0	0.00	17	2.13
Twenty-ninth through thirty-first year of C.O.	0.000	0	0	0.00	9	3.00	0.000	0	0	0.00	5	1.67
Thirty-second through ninety-ninth year of C.O.	0.500	1	2	4.04	5	3.98	0.500	1	0	0.00	0	0.00
Total	18.830		40	2.12	2347	5.72	37.660		48	1.27	3205	4.02

[a]Age is defined to be the time (in years) from the start of commercial operation to the time of the shutdown event, except for the first line, which lists reactors not yet in commercial service (see b below).
[b]This category includes reactors licensed for full-power operation but not yet in commercial operation. During this reporting period reactors in this category included 1 BWR (Shoreham) and no PWRs.

U.S. Nuclear Regulatory Commission Information and Analyses

Edited by D. E. Hickman

Reactor Coolant System Blowdown at Wolf Creek on September 17, 1994

By John V. Kauffman and Sanford L. Israel[a]

Abstract: *On September 17, 1994, an inadvertent blowdown occurred at Wolf Creek reactor; about 34 822 L (9 200 gal) of reactor coolant passed through the residual heat removal (RHR) system to the refueling water storage tank (RWST) while the Wolf Creek reactor was shut down in Mode 4 on RHR cooling [2.5 MPa and 149 °C (350 psig and 300 °F)]. This event occurred because of concurrent activities involving manipulations of RHR valves while cooling down to begin a refueling outage. The inadvertent blowdown of reactor coolant was terminated in about a minute by closing one of the RHR valves that was being manipulated. Continued blowdown through the RHR system would have uncovered the reactor hot leg and introduced steam into the RWST header line, which is the water supply line for the emergency core cooling system (ECCS) pumps. The Nuclear Regulatory Commission Office for Analysis and Evaluation of Operational Data performed an event review to provide better understanding of the event initiation; operator response; potential engineering issues; and possible event progression without the initial, successful operator intervention.*

This article describes the plant conditions prior to the September 17, 1994, blowdown of reactor coolant at Wolf Creek reactor; initiation of the blowdown; the blowdown itself and operator response to it; and the results of the Nuclear Regulatory Commission (NRC) analysis of the human performance and engineering aspects of the event. This analysis was issued as an NRC AEOD Special Study, S95-01, "Reactor Coolant System Blowdown at Wolf Creek on September 17, 1994," in March 1995.

The Wolf Creek event disclosed a previously unrecognized design vulnerability: a piping arrangement whose inappropriate use while on residual heat removal (RHR) cooling could result in a fast loss-of-coolant event and a consequential common-mode loss of emergency core cooling system (ECCS) mitigation capability if an extended blowdown occurred through this path. The mitigation of an extended blowdown if the ECCS pumps have failed is uncertain. Uncertainties that affect a conditional core damage probability calculation for this sequence of events depend largely on values used for operator actions, uncertainties about common-mode impairment of ECCS equipment that takes suction from the refueling water storage tank (RWST) header, and the initiation of reflux cooling. The failure to control work activities resulted in the initiation of the event, which preliminary review indicates will be among the more significant ones of recent years from a safety standpoint.

EVENT NARRATIVE DESCRIPTION

Initial Plant Conditions

Shortly after 4:00 a.m. on Saturday, September 17, 1994, Wolf Creek was shut down in Mode 4, cooling down at the beginning of Refueling Outage VII. The reactor coolant system (RCS) was at about 2.5 MPa and

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149 °C (350 psig and 300 °F). Two reactor coolant pumps (RCPs) were secured at least 8 h before the event. The steam generators were filled, but the condenser and condensate systems were secured about 2 h before the event. The cold overpressure protection system was armed 8 h prior to the event. The safety injection (SI) pumps, one of two centrifugal charging pumps, and the positive displacement pump (PDP) were secured and breakers opened as part of the cold overpressure protection.

About 4 h earlier, RHR train A was placed in service to cool the reactor. About 25 MW(t) of decay heat was being removed by RHR train "A" [10 371 L/min (2 740 gal/min)] with a mixed outlet temperature of 112 °C (234 °F) and inlet temperature of 150 °C (302 °F). Auxiliary feedwater was available.

The control room (CR) operators were busy. A second relief crew consisting of licensed and nonlicensed operators augmented the on-shift crew. Several activities were in progress, and several distractions occurred during the shift.

Prior to the outage, a chemistry sample analysis determined that the "B" RHR train boron concentration (about 1200 ppm) was lower than RCS boron concentration (about 2000 ppm). This lower concentration was attributed to check valve backleakage at the RCS pressure boundary. The procedure for start-up of an RHR train required RCS and RHR boron concentrations to be within 50 ppm, which necessitated raising the boron concentration of the "B" RHR train by recirculation to the RWST.

Earlier in the shift, at 9:25 p.m. on September 16, 1994, and again at about 3:00 a.m. on September 17, 1994, the shift supervisor (SS) held discussions with maintenance personnel involved with the retest of HV-8716A (see Fig. 1, simplified diagram) (RHR train "A" isolation valve in the crossover line to hot-leg recirculation loops 2 and 3). The SS granted permission to adjust the packing of HV-8716A, which would require stroking this valve to conduct valve testing, provided appropriate plant conditions existed as determined by the on-shift supervising operator (SO).

Reactor Coolant System Blowdown

Just prior to the event, the CR operators were deployed as follows:

- The on-shift SS was in his office performing administrative duties, while the shift SO was at his desk keeping the CR log and monitoring plant activities.
- The on-shift reactor operator (RO) was controlling the chemical and volume control system to raise the pressurizer level slowly in preparation for taking the RCS solid. This activity was complicated by a malfunctioning nitrogen regulator on the volume control tank (VCT).
- The on-shift balance of plant (BOP) operator was involved with aligning the "B" RHR train for recirculation to the RWST to increase boron concentration in the "B" RHR train. In addition, the operator tracked and occasionally compensated for sluicing between the component cooling water trains.
- One nuclear station operator (NSO) had discussed with the BOP operator the "B" RHR train lineup for recirculation to increase its boron concentration. This NSO was proceeding to BN 8717 (RHR pump return to RWST valve) with instructions to open it slowly in accordance with the procedure.
- A second RO was plotting the pressurizer cooldown rate, and a third, who had recently returned from adjusting the VCT pressure regulator, was controlling the "B" diesel generator (DG) 24-h run.
- The relief crew SO had been assisting the on-shift SO and was now at the radiation monitor panel involved with a surveillance, and the relief crew SS was standing near the feedwater system control panel.

An electrician informed the BOP operator that the packing adjustment on valve HV-8716A had been completed and requested that it be stroked for the valve test. The BOP operator conferred with the on-shift SO and received concurrence to conduct the stroke test. Meanwhile, the NSO had arrived in the valve room that contained valves HV-8716A and BN 8717. The electrician and the NSO were about 1 m (3 ft) apart, engaged in different evolutions. The NSO was going to open valve BN 8717 manually as part of the boration of the "B" RHR loop, whereas valve HV-8716A was going to be stroked open and closed from the CR.

In the CR, the BOP operator stroked HV-8716A for the first time. About 30 s later, the BOP operator pushed the open button to start the second stroke test at about the same time that BN 8717 was fully opened. Opening HV-8716A concurrently with BN 8717 created a flow path to blow down the RCS to the RWST.

The NSO noted flow noise when valve BN 8717 was initially cracked off its seat. He interpreted this as pressure equalization across the valve, which he expected. He also heard a loud noise like a water hammer. He then proceeded to open the valve slowly. At about the time the valve was fully open, he and the electrician heard a loud water hammer. The NSO hurried to the plant page and reported what he had heard; CR personnel instructed him to close BN 8717. The first time valve HV-8716A was stroked open and then shut, BN 8717 was likely closed or cracked open, allowing only a small flow path. The second time valve HV-8716A was opened, a large flow path was created from the RCS to the RWST through the RHR system because valve BN 8717 was open.

Meanwhile, the first annunciator received in the CR was the RWST high-level alarm. The on-shift RO saw that the pressurizer high-level annunciator was clear and checked pressurizer level. Observing that the pressurizer level trend recorder and the hot calibrated pressurizer level instruments were pegged low, he announced the loss of pressurizer level.

The on-shift SO ordered securing of the RCPs, maximizing charging from the centrifugal charging pump and isolating low-pressure letdown. Meanwhile, the relief crew SO proceeded to the RHR control board area where the BOP operator was standing. The relief crew SO observed the open indication on valve HV-8716A and asked the BOP operator if valve BN 8717 was open. When the BOP operator responded that it was, the relief crew SO understood and identified the flow path for reactor coolant through HV-8716A and BN 8717 to the RWST. The relief crew SO then informed the BOP operator that valve HV-8716A should be closed. The BOP operator closed HV-8716A, isolating the flow path and terminating the uncontrolled blowdown, which lasted 66 s.

The operators proceeded to recover pressurizer level in a slow, controlled manner to minimize the thermal stresses on the pressurizer surge line. After the plant was stabilized, most activities were stopped until the situation could be assessed. The operators did not refer to the shutdown loss-of-coolant accident (LOCA) procedure; however, some alarm procedures were reviewed following the event.

The licensee estimated that about 34 822 L (9 200 gal) drained out of the pressurizer from the RCS to the RWST through a 20-cm (8-in.) line in 66 s. This filled the RWST and overflowed approximately 2 460 L (650 gal) through the installed piping to the radioactive waste holdup tank (RHUT). This estimate was based on recorded water level measurements in the pressurizer. The RCS pressure went from 2.5 MPa (350 psig) to 1.65 MPa (225 psig) based on CR indications, not recorded data. The flow along the RHR discharge line to the RCS went to zero, which indicated that RHR flow was diverted to the RWST header line.

RESIDUAL HEAT REMOVAL SYSTEM OPERATION

The RHR system (Fig. 1) is composed of two essentially identical trains that operate similarly. When the system is in RHR cooling mode, water is drawn from an RCS hot leg to the RHR pump, which discharges to two parallel lines. One line contains the RHR heat exchanger (HX), and the other is a bypass line with an automatically controlled regulating valve. Flow through the RHR HX line is manually set to maintain an acceptable cooldown rate.

The crossover line between the two RHR trains contains two isolation valves, EJ HV-8716A and EJ HV-8716B, that are normally closed while using RHR cooling. A 20-cm (8-in.) RHR-RWST discharge line connects the RHR crossover line to the common header line from the RWST. This RHR-RWST line contains a single manual valve (BN 8717) that is "locked" closed. The RHR-RWST line connects to the RHR crossover line between valves HV-8716A and HV-8716B. A common SI line to two RCS hot legs also connects between valves HV-8716A and HV-8716B. At power, valves HV-8716A and HV-8716B in the RHR crossover line are normally open.

Because of a valving error, a blowdown pathway was established when the RCS hot leg was connected to the common header line from the RWST. Introduction of hot water and steam had the potential to disable the ECCSs via steam voids or net positive suction head limitations in the common suction header.

HUMAN PERFORMANCE ASPECTS OF THE EVENT

Task Involvement and Awareness

A licensee report documented the following conclusions regarding a number of human factors contributing to the initiation of this event:

• "Two activities, governed by SYS EJ-120 [RHR system operating procedure] and WR 05811-94 [work request] were performed simultaneously. These activities are incompatible with each other because SYS EJ-120 uses EJ HV-8716A as a 'boundary valve' for reactor coolant."

• "The BOP operator did not take the time to perform an adequate brief, review [the] procedure, or review the prints prior to performing SYS EJ-120 for borating the B RHR train. Also, he did not do an adequate job of

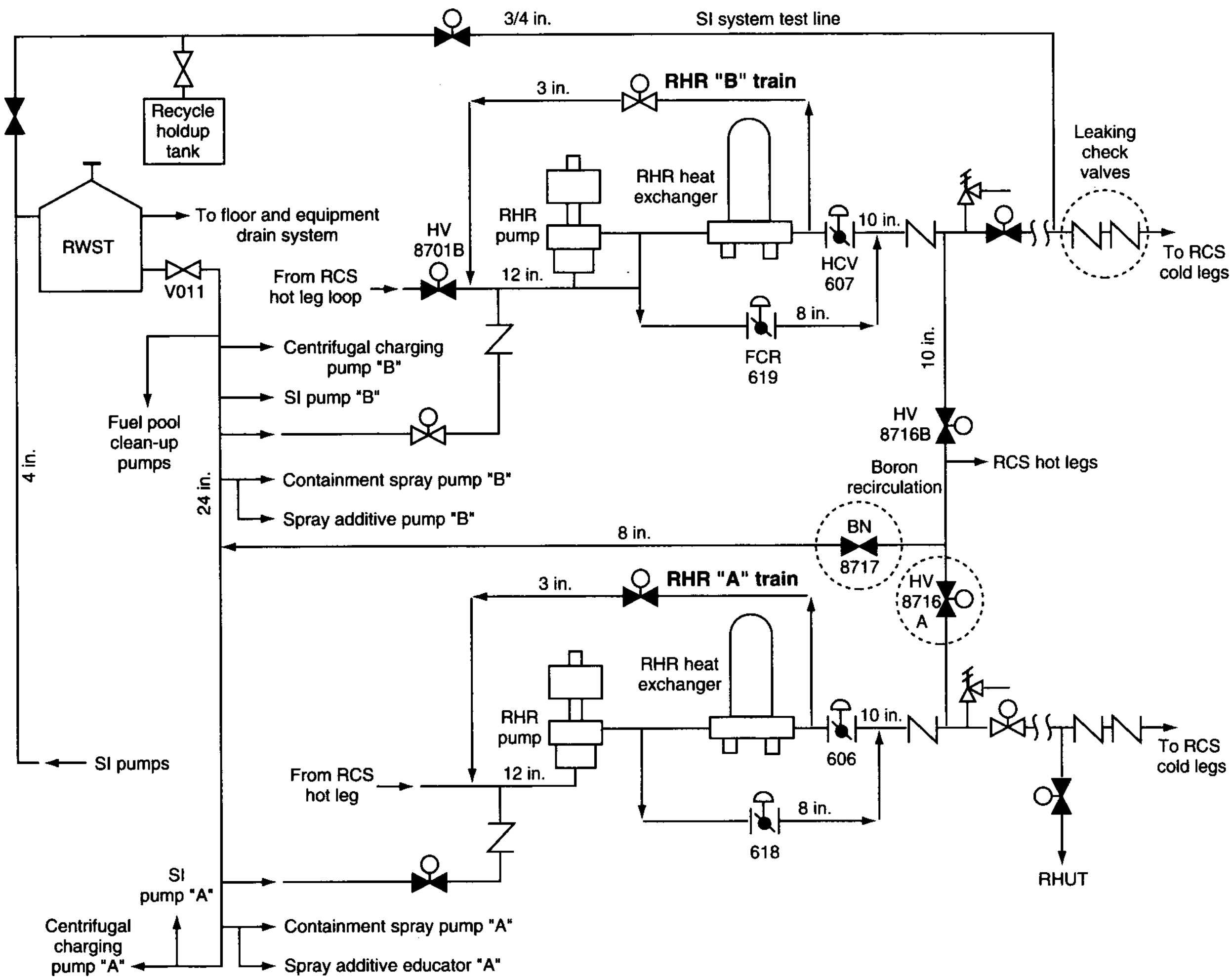

Fig. 1 Valve lineup before event. (RCS is reactor coolant system, SI is safety injection, RWST is refueling water storage tank, RHR is Residual Heat Removal, and RHUT is radioactive waste holdup tank.)

STAR [Stop-Think-Act-Review—self-checking] when the request to stroke EJ HV-8716A was made."

• "The [on-shift] SO did not exercise proper command and control techniques to maintain full awareness of plant conditions. The SO authorized performance of SYS EJ-120 concurrent with the stroking of EJ HV-8716A while the A RHR train was providing cooling for the RCS."

Work Controls

The outage planning process at Wolf Creek places heavy reliance on the CR crew to identify potential problems. This reliance, in conjunction with numerous CR activities, contributed to initiation of this event. The licensee did have an emergent work control process, but it was not used for planning the retest of valve HV-8716A.

Initial and Subsequent Operator Response

The operating staff diagnosed the problem and took actions that stopped the event. A relief shift SO (not involved with the evolutions that led to the blowdown) identified the flow path that was causing the rapid pressurizer level decrease. Thus "fresh eyes" can often identify and correct human errors. Diagnosing the symptoms and identifying the blowdown path were knowledge-based responses.

The operators did not refer to procedures during their initial response to the event. Initially, operators tripped running RCPs, maximized charging flow, and isolated letdown. Hence their initial response relied on their training and their knowledge of general actions to be taken or rules governing reactions to a rapid loss of pressurizer level or LOCA event, in particular a LOCA in Modes 1, 2, and 3.

After the blowdown was stopped, the operators referred to alarm response procedures. Their subsequent plant recovery was based on various considerations and requirements such as technical specifications (TS), concerns for pressurizer surge line thermal stresses, and the ongoing test run of the "B" DG. Some important actions, such as emergency classification and declaration, were not considered, at least partially because the applicable procedure was neither entered during the transient nor checked after the plant was stabilized.

The licensee's review concluded that all personnel actions in response to the event were appropriate; however, emergency action levels should have been consulted immediately after the event. The licensee's review of the event also concluded that no emergency classification was warranted for this event.

Procedures and Their Use

Shutdown LOCA Procedure (OFN BB-031). Wolf Creek had an off-normal procedure, OFN BB-031, "Shutdown LOCA," that was intended for situations like this event. OFN BB-031 was formatted similarly to the Wolf Creek emergency operating procedures and was comprehensive—it contained 143 pages with 81 steps and 5 appendixes (about half of the pages were an identical continuous action page provided for operator ease of use). One of the symptoms for entry was an "uncontrolled decrease in PZR [pressurizer] level" during Modes 3, 4, or 5. OFN BB-031 was based on Westinghouse Owners Group (WOG) guidelines for a shutdown LOCA. The operating crew had received training on a shutdown LOCA scenario and other shutdown scenarios immediately before the plant shutdown.

Some of the operator actions directed by OFN BB-031 differed from the actions of the operators during the event; for example, step 2 and a foldout page both direct that, if any RHR pumps are taking suction from the RCS and pressurizer level is less than 4%, then the RHR pumps are to be stopped and placed in pull-to-lock. According to the licensee's bases document, the purpose of this step is to prevent damage to the pumps and allow for future pump operation. During the actual event the RHR pumps were not tripped. Leak identification and isolation are included in step 10, which describes what to do if pressurizer level has been restored or is greater than 4%. Another foldout page step helps determine the emergency classification level. On the basis of interviews, the on-shift SS did not consider making an emergency classification.

The diagnosis of the flow path by the relief crew SO and subsequent isolation terminated the event prior to loss of core cooling. The relief crew SO's engaging in diagnostic activities appears appropriate; he was not on shift and was not responsible for directing or supervising the implementation of the operator response to the blowdown. The on-shift crew did not implement the applicable procedure. The rapidity of the inventory loss, rather than a conscious decision, appears to be the reason why operators did not use the procedure. On the basis of interviews, the crew felt that the event was terminated and the plant stabilized once HV-8716A was closed; so referring to the procedure was not thought to be required.

Loss of RHR Cooling Procedure. A 150-page document, "Loss of RHR Cooling" (OFN EJ 015), developed in 1990 in response to Generic Letter 88-17, "Loss of Decay Heat Removal,"[1] reflects guidance developed by the WOG. Its entry conditions included loss of RHR flow, erratic RHR pump current, and erratic RHR flow oscillations. This procedure directed actions to recover RHR by stopping the pumps, refilling the RCS, and venting the RHR pumps. Near the end of the procedure, directions were given to use alternate heat removal methods.

Usability of Procedures. A review of several procedures related to this event raised questions about their usability; for example, the shutdown LOCA procedure, OFN BB-031, has 5 continuous-action statements on the left-hand page, and at least 17 other "check" steps appear within the body of the procedure, 2 of which are also continuous action. Similarly, the procedure for starting an RHR train (SYS EJ-120) contains 15 precautions and limitations at the beginning of the procedure and another 34 notes and cautions in the 46-page body. Some of these precautions and notes appear to be continuous-action-type statements.

In regard to OFN BB-031, certain critical actions, such as cold overpressure protection and tripping the RCPs, might not be implemented in a timely fashion while following this procedure. Operators tripped the running RCPs during the September 17 event. The licensee is modifying OFN BB-031 following the event. Planned changes include directions to trip the RCPs immediately for a rapid depressurization, enhancements to the RCP-tripping criteria, and enhancements to the SI reduction criteria for cold overpressurization or pressurized thermal shock (PTS) concerns.[2] The licensee plans further evaluations of the mitigation strategy of the procedure.

Licensee analyses subsequent to the event showed that, under some initial conditions, the operators may have only 3 to 5 min to isolate the blowdown path before steam in the common suction piping could degrade or fail SI, centrifugal charging, and RHR pumps. Thus, for some initial conditions, timely leak isolation could be very important. Leak isolation, however, is not the principal mitigation strategy in the applicable procedure. The applicable procedure, if used, would not have directed leak isolation within the time needed to prevent potential failure of ECCS pumps. The licensee offered reasons why leak isolation is not the principal mitigation strategy (e.g., isolation of RHR defeats low-temperature overpressure protection, and concerns exist about the ability of valves to be reopened to use RHR for cooling).

Operational Experience

The licensee identified three previous events, including one at Wolf Creek in 1983, similar to this event. According to the licensee, a 1990 Braidwood event most likely resulted in the placement of an operator aid in the CR at Wolf Creek that shows the location of valve BN 8717.

In the United States, in 1200 pressurized-water reactor years, at least 19 related loss-of-coolant events have occurred with varying blowdown rates while the reactor was on RHR cooling. Boiling and two-phase flow were not issues for most of these 19 events, which were identified in different studies related to shutdown cooling and do not represent an exhaustive search for data. In most cases, the flowpath was from the RCS hot leg through the RHR system back to the RWST via some common discharge line. In most plant designs, this discharge line is not connected to the RWST header line (ECCS suction line) as it was at Wolf Creek. The coolant loss was terminated when an operator closed a valve in the majority of these events. In a 1989 Braidwood event, however, the operator quickly isolated one of the RHR trains, but the 238 000-L (63 000-gal) loss continued over 2 h because the wrong train was isolated. For most events, temperatures less than 93 °C (200 °F) reduced the potential exposure to complications associated with boiling and two-phase flow.

Compressed Outage Schedule

On the basis of interviews with the licensee as well as the licensee's investigation of the event, several observations can be made. The additional work activities and workers involved in these activities likely contributed to a higher cognitive load for the on-shift crew that may have made the task of maintaining the "big picture" more difficult.

The compressed refueling outage schedule was several weeks shorter than previous outages at Wolf Creek. The amount of ongoing work during the shutdown and cooldown of the reactor prior to the outage was higher than typically experienced during other shutdowns proceeding to refueling. The crews expressed the opinions that work activities were well controlled and coordinated and that the extra workload was not a significant problem. Nonetheless, the lack of control of multiple work activities affected plant configuration control, which allowed the rapid blowdown of the RCS.

At Wolf Creek, one of the Operations Outage Supervisors who reviewed the schedule was concerned about the potential to discharge the RCS to the RWST. This concern was communicated to Outage Management and the SS on September 14, 1994. Positive means (such as equipment tagging) were not used to keep these activities separate. Thus the final decision to perform testing of HV-8716A rested with the operating crew SS and the SO and their "comfort levels."

ENGINEERING AND OPERATIONAL CONSIDERATIONS

During NRC review of the event, several engineering and operational considerations became apparent that have relevance to the successful mitigation of a hypothetical extended blowdown.

Thermal–Hydraulic Response

The mixed mean temperature of the water going to the RWST header line is a function of the flow split and the heat-transfer characteristics of the RHR HX. No RHR discharge temperatures were measured during the 66-s transient because the temperature transmitter is located next to the downstream flow orifice that lost flow during the transient. At the end of the transient, a temperature of 127 °C (261 °F) was recorded, presumably the mixed mean RHR temperature at the end of the transient.

The recorded 127 °C (261 °F) water temperature is near the saturation temperature of water in the horizontal RWST line [about 16.8 m (55 ft) below the surface of the water in the RWST]. The ECCS pumps, located 3 to 5.5 m (10 to 18 ft) below the RWST line, require 4.9 to 6.1 m (16 to 20 ft) of net positive suction head to preclude cavitation. After the event, the licensee stated that no assurance existed that the ECCS pumps would fulfill their function while drawing water from the RWST following the event.

NRC's initial concern about this event was that an unabated blowdown through the RHR system would have uncovered the reactor hot leg and introduced steam into the RWST header line, which would potentially disable the only source of water for all the ECCS pumps needed to mitigate a LOCA.

NRC performed simulation of the Wolf Creek event with an unabated blowdown using RELAP5 and a Seabrook plant layout. The 34 822-L (9 200-gal) blowdown in 66 s was approximated by a 0.01-m^2(0.1-ft^2)- or 10.7-cm(4.2-in.)-diameter hole in the bottom of a hot-leg pipe. This approximation was necessary because the RHR and RWST piping systems are not currently incorporated in the RELAP5 model. Two cases were run, with RCPs on and off. As expected, the vessel inventory transient for these cases was more benign than the analysis of the 15.2-cm (6-in.) break in a 4-loop plant analyzed in WCAP-12476, "Evaluation of LOCA During Mode 3 and Mode 4 Operation for W NSSS."

These calculations show a two-phase mixture in the hot leg starting at about 3 min. More than 30 min elapsed before core uncovery with the RCPs running. Even more time is available if the pumps are tripped. These time frames are uncertain, however, because the model did not account for two-phase pressure losses in the RHR system and the 61-cm (24-in.) RWST piping.

The licensee had Westinghouse Electric Corporation (W) perform thermal–hydraulic calculations to examine the conditions in the RWST header line if the blowdown had continued unabated. Review by the licensee indicated that analyses are very sensitive to nuances in the piping configuration. The licensee indicated that a revised W analysis showed a 90% void fraction in the RWST header line starting at 6 min and continuing until the blowdown path is isolated.[3] Under these conditions, the multistage SI pumps, which take suction from this line, would be expected to fail if operated. The potential mitigation of an extended blowdown under these adverse conditions is undetermined from phenomenological and human factors standpoints. If the blowdown path were not isolated, the licensee estimated that the core uncovery would begin in 30 min.

The licensee stated that the high-pressure pump manufacturer had estimated the pumps would last only 1.5 min if steam bound. The licensee also noted that voids in the RHR system at about 3.5 min create concerns about RHR pump operability because of vapor collapse and water hammer during RHR pump restart.

Use of Blowdown Mitigation Procedures

Which procedure the operators would open given an extended blowdown is unknown. A successful recovery from an unabated blowdown without ECCS pumps is not certain because of ambiguities in the procedures and questions about operator actions.

Procedure OFN BB-031, "Shutdown LOCA," would isolate the RHR loop and align it for injection at step 28. If the RHR-RWST discharge line is not isolated, however, the low-pressure RHR flow (if recovered) would still be directed to the RWST header and would not reach the RCS. If the RHR-RWST line is isolated, some of the ECCS pumps may be recoverable, depending on the prior operator action to activate these pumps as well as the pumps' survivability. Furthermore, all the pumps may not vapor bind because the ECCS pumps are started one at a time, the high-pressure pumps draw water from the bottom of the RWST header line, and the blowdown and pumping flow rates are relative.

At step 31 in OFN BB-031, direction is given to use the steam generators and the atmospheric relief valves as a heat sink if the hot-leg temperatures are not stable. This path is the most promising if the RCS is isolated. In the RELAP5 analysis, however, the hot-leg temperatures stay fairly stable if the coolant loss path is not isolated. At step 66, the operator is directed to the PDP, which could be used for charging flow if the centrifugal pumps are not operating. The PDP flow rate, however, is less than the decay heat boil-off rate. The operability of the charging pumps and the PDP is undetermined because the charging pump connected to the VCT had been switching to the RWST header line prior to the event because of other problems during the shutdown. Another concern is that the operators would become distracted when the ECCS pumps started failing and would try to restore failed pumps.

The licensee estimated that performing RHR pump venting would take 10 to 15 min if the pumps become vapor bound. Under better circumstances with coolant temperatures less than saturation, however, restoring

RHR cooling at Waterford took 3.5 h even with a ventable system.[4]

Alternatively, the operators could have been in OFN EJ-15, "Loss of RHR Cooling," which is primarily concerned with recovering the RHR system in the cooling mode. The isolation of the coolant loss path is directed at step 40. This procedure directs the use of the steam generators at step 43 for heat removal. It also activates the accumulators at step 70 at the end of the procedure. The RCS pressures at that time may preclude use of the accumulators.

Residual Heat Removal System WATER Hammer

The causes of the apparent water hammers heard during the event were not determined; however, questions of adverse effects raised by the water hammer issue include the following:

- What would happen if the blowdown progressed and steam came into contact with cold water in the RWST?
- What would happen when steam condenses in the RHR HX?
- Can excessive pressure pulses occur in the RHR system if the operator terminates the high initial blowdown rate quickly?

Boron Concentration Variances

The boron evolution was precipitated by stringent concentration requirements in the procedures. At the time of the event, procedure SYS EJ-120, "Startup of Residual Heat Removal Train," required that each train be sampled prior to being put into operation to ensure that the boron concentration is within 50 ppm of the concentration in the RCS, which is being borated continuously during shutdown. Train "A" was sampled, found to have a boron concentration greater than 2400 ppm, and put into service about 4 h before the event. Train "B" was sampled while the reactor was in Mode 3 and was found to have a concentration of 1230 ppm. The licensee considered borating the "B" RHR train prior to the outage; however, the TS prohibits closing the cross-tie valves, HV-8716A and HV-8716B, in Modes 1 and 2.

The licensee determined subsequently that the boron concentration in the "B" train would not cause a criticality problem even if introduced unmixed into the reactor core. To minimize the need to establish the system lineup that led to this event, the licensee has changed the boron requirements for putting an RHR train into service:

- If the concentration meets the minimum shutdown margin for boron concentration, operation of the RHR train is acceptable without additional action.
- If the boron concentration is less than 100 ppm lower than that required by the minimum shutdown margin and *two RCPs are operating,* operation of the RHR train is acceptable without additional action.
- For all other situations, the RHR train must be borated before use.

Check Valve Leakage

A contributing factor to the event was the check valve back leakage from the RCS into the RHR system while the plant was at power because this reduced the boron concentration in RHR Train B.

The leakage needed to dilute the boron concentration in an RHR train is quite low. A leakage rate of 0.038 L/min (0.01 gal/min) would displace the initial water inventory in an RHR train over 1 year. If the leakage rate is 0.38 L/min (0.1 gal/min) (less than TS limits on RCS leakage), the water inventory turnover could be accomplished in about 1 month. Thus, obtaining very low boron concentrations in an RHR train at the end of an operating cycle is possible. To dilute an RHR train, this leakage has to be past the third check valve from the RCS. This check valve is not leak tested during every refueling.

CONCLUSIONS

The following conclusions are based on a review of the event and information relevant to a potential extended blowdown if the problem had not been isolated quickly:

- Unrecognized Design Vulnerability

  The Wolf Creek event disclosed a previously unrecognized design vulnerability: a piping arrangement connecting the discharge of both trains of RHR to the RWST header line whose inappropriate use while on RHR cooling could result in a fast loss-of-coolant event and a consequential common-mode loss of ECCS mitigation capability if an extended blowdown occurred through this path.

- Control of Work Activities

  Operators failed to control work activities appropriately, and this failure resulted in the initiation of the event. Many factors affected operators' ability to control work activities.

- Initial Response

  The operating staff diagnosed the blowdown and closed a valve, which stopped the event.

- Mitigation of an Extended Blowdown

  The mitigation of an extended blowdown if the ECCS pumps are failed is undetermined. Uncertainties that affect a conditional core damage probability calculation for the Wolf Creek sequence of events depend largely on values used for operator actions, uncertainties about common-mode impairment of ECCS equipment that takes suction from the RWST header, and the initiation of reflux cooling. Preliminary review indicates the event is among the most significant events of recent years from a safety standpoint.

- Safety Significance of Design Vulnerability

  The potential safety significance of the design vulnerability was not fully understood or appreciated initially.

REFERENCES

1. U.S. Nuclear Regulatory Commission, Generic Letter 88-17, *Loss of Decay Heat Removal,* October 17, 1988.
2. Wolf Creek Nuclear Operating Corporation, Licensee Event Report 50-482/94-13, *Personnel Error Resulted in an Unanticipated Loss of Reactor Coolant Level, Wolf Creek Nuclear Generating Station,* January 4, 1995.
3. G. Neises, Wolf Creek Generating Station, memorandum to L. Stevens, Wolf Creek Generating Station, *RCS Drain Down Event—Analysis Summary Report,* March 1, 1995.
4. Louisiana Power and Light, Licensee Event Report 50-382/86-15, *Simultaneously Using Two Methods of Draining Reactor Coolant System Results in Loss of Shutdown Cooling, Waterford Generating Station Unit 3,* August 13, 1986.

Recent Developments

Edited by M. D. Muhlheim

Reports, Standards, and Safety Guides

By D. S. Queener

This article contains four lists of various documents relevant to nuclear safety as compiled by the editor. These lists are: (1) reactor operations-related reports of U.S. origin, (2) other books and reports, (3) regulatory guides, and (4) nuclear standards. Each list contains the documents in its category which were published (or became available) during the April 1995 through September1995 reporting period. The availability and cost of the documents are noted in most instances.

OPERATIONS REPORTS

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

NRC Office of Nuclear Reactor Regulation

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

NRC Generic Letters

NRC GL 89-04, Supplement 1 *Guidance on Developing Acceptable Inservice Testing Programs,* April 4, 1995, 3 pages plus 3 pages of attachments.

NRC GL 95-03 *Circumferential Cracking of Steam Generator Tubes,* April 28, 1995, 4 pages plus one-page attachment.

NRC GL 95-04 *Final Disposition of the Systematic Evaluation Program, Lessons-Learned Issues,* April 28, 1995, 13 pages plus one-page attachment.

NRC GL 95-05 *Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking,* August 3, 1995, 7 pages plus 26 pages of attachments.

NRC GL 95-06 *Changes in Operator Licensing Program,* August 15, 1995, 8 pages plus one-page attachment.

NRC GL 95-07 *Pressure Locking & Thermal Binding of Safety-Related Power-Operated Gate Valves,* August 17, 1995, 14 pages plus one-page attachment.

NRC Information Notices

NRC IN 95-21 *Unexpected Degradation of Lead Storage Batteries,* April 20, 1995, 3 pages plus one-page attachment.

NRC IN 95-22 *Hardened or Contaminated Lubricants Cause Metal-Clad Circuit Breakers Failures,* April 21, 1995, 4 pages plus one-page attachment.

NRC IN 95-23 *Control Room Staffing Below Minimum Regulatory Requirements,* April 24, 1995, 3 pages plus one-page attachment.

NRC IN 95-24 *Summary of Licensed Operator Requalification Inspection Program Findings,* April 25, 1995, 3 pages plus 3 pages of attachments.

NRC IN 95-25 *Valve Failure During Patient Treatment with Gamma Stereotactic Radiosurgery Unit,* May 11, 1995, 3 pages plus one-page attachment.

NRC IN 95-26 *Defect in Safety-Related Pump Parts Due to Inadequate Heat Treatment,* May 31, 1995, 2 pages plus 12 pages of attachments.

NRC IN 95-27 *NRC Review of Nuclear Energy Institute, "Thermo-Lag 330-1 Combustibility Evaluation Methodology Plant Screening Guide,"* May 31, 1995, 2 pages plus 8 pages of attachments.

NRC IN 95-28 *Emplacement of Support Pads for Spent Fuel Dry Storage Installations at Reactor Sites,* June 5, 1995, 4 pages plus one-page attachment.

NRC IN 95-29 *Oversight of Design and Fabrication Activities for Metal Components Used in Spent Fuel Dry Storage Systems,* June 7, 1995, 3 pages plus 3 pages of attachments.

NRC IN 95-30 *Susceptibility of Low-Pressure Coolant Injection and Core Spray Injection Valves to Pressure Locking,* August 3, 1995, 4 pages plus one-page attachment.

NRC IN 95-31 *Motor-Operated Valve Failure Caused by Steam Protector Pipe Interference,* August 9, 1995, 3 pages plus 2 pages of attachments.

NRC IN 95-32 *Thermo-Lag 330-1 Flame Spread Test Results,* August 10, 1995, 2 pages plus one-page attachment.

NRC IN 95-33 *Switchgear Fire and Partial Loss of Offsite Power at Waterford Generating Station, Unit 3,* August 23, 1995, 4 pages plus one-page attachment.

NRC IN 95-34 *Air Actuator and Supply Air Regulator Problems in Copes-Vulcan Pressurizer Power-Operated Relief Valves,* August 25, 1995, 4 pages plus 2 pages of attachments.

NRC IN 95-35 *Degraded Ability of Steam Generators to Remove Decay Heat by Natural Circulation,* August 28, 1995, 3 pages plus one-page attachment.

NRC IN 95-36 *Potential Problems with Post-Fire Emergency Lighting,* August 29, 1995, 3 pages plus 3 pages of attachments.

NRC IN 95-37 *Inadequate Offsite Power System Voltages During Design-Basis Events,* September 7, 1995, 4 pages plus one-page attachment.

NRC IN 95-38 *Degradation of Boraflex Neutron Absorber in Spent Fuel Storage Racks,* September 8, 1995, 3 pages plus one-page attachment.

NRC IN 95-39 *Brachytherapy Incidents Involving Treatment Planning Errors,* September 19, 1995, 4 pages plus one-page attachment.

NRC IN 95-40 *Supplemental Information to Generic Letter 95-03, "Circumferential Cracking of Steam Generator Tubes,"* September 20, 1995, 3 pages plus one-page attachment.

NRC IN 95-41 *Degradation of Ventilation System Charcoal Resulting from Chemical Cleaning of Steam Generators,* September 22, 1995, 3 pages plus 2 pages of attachments.

NRC IN 95-42 *Commission Decision on the Resolution of Generic Issue 23, "Reactor Coolant Pump Seal Failure,"* September 22, 1995, 2 pages plus one-page attachment.

NRC IN 95-43 *Failure of the Bolt-Locking Device on the Reactor Coolant Pump Turning Vane,* September 28, 1995, 2 pages plus one-page attachment.

NRC IN 95-44 *Ensuring Compatible Use of Drive Cables Incorporating Industrial Nuclear Company Ball-Type Male Connectors,* September 26, 1995, 2 pages plus 2 pages of attachments.

Other Operations Reports

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

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

NUREG-0090, Vol. 17, No. 4 *Report to Congress on Abnormal Occurrences for October–December 1994,* May 1995, 28 pages (GPO).

NUREG-0090, Vol. 18, No. 1 *Report to Congress on Abnormal Occurrences for January–March 1995,* July 1995, 15 pages (GPO).

NUREG-1423, Vol. 5 *A Compilation of Reports of The Advisory Committee on Nuclear Waste, July 1993–June 1995,* August 1995, 65 pages (GPO).

NUREG-1525 *Assessment of the NRC Enforcement Program,* J. Lieberman et al., April 1995, 170 pages (GPO).

NUREG-1526 *Lessons Learned from Early Implementation of The Maintenance Rule at Nine Nuclear Power Plants,* C. D. Petrone et al., June 1995, 35 pages (GPO).

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

NUREG/CR-2907, Vol. 13 *Radioactive Materials Released from Nuclear Power Plants, Annual Report 1992,* J. Tichler et al., Brookhaven National Lab., N.Y., August 1995, 350 pages (GPO).

NUREG/CR-3469, Vol. 8 *Occupational Dose Reduction at Nuclear Power Plants: Annotated Bibliography of Selected Readings in Radiation Protection and ALARA,* S. G. Sullivan et al., Brookhaven National Laboratory, N.Y., May 1995 (GPO).

NUREG/CR-5758, Vol. 5 *Fitness for Duty in the Nuclear Power Industry. Annual Summary of Program Performance Reports CY 1994,* M. Hattrup et al., Pacific Northwest Labs., Wash., August 1995, 80 pages (GPO).

NUREG/CR-6016 *Aging and Service Wear of Air-Operated Valves Used in Safety-Related Systems at Nuclear Power Plants,* D. F. Cox et al., Oak Ridge National Lab., Tenn., May 1995, 65 pages (GPO).

NRC Office for Analysis and Evaluation of Operational Data

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

NUREG-1275, Vol. 11 *Operating Experience Feedback Report—Turbine-Generator Overspeed Protection Systems, Commercial Power Reactors,* H. L. Ornstein, April 1995, 95 pages (GPO).

NUREG-1527 *NRC's Object-Oriented Simulator Instructor Station,* J. I. Griffin and J. P. Griffin, June 1995, 100 pages (GPO).

DOE- and NRC-Related Items

NUREG-0383, Vol. 2, Rev. 18 *Directory of Certificates of Compliance for Radioactive Materials Packages. Report of NRC-Approved Packages,* October 1995, 560 pages (GPO).

NUREG-1482 *Guidelines for Inservice Testing at Nuclear Power Plants,* P. Campbell, April 1995, 150 pages (GPO).

NUREG-1600 *General Statement of Policy and Procedures for NRC Enforcement Actions. Enforcement Policy,* July 1995, 25 pages (GPO).

NUREG/CP-0140, Vols. 1–3 *Proceedings of the U.S. Nuclear Regulatory Commission Twenty-Second Water Reactor Safety Information Meeting,* October 24–26, 1994, Bethesda, Md., S. Monteleone, April 1995, 1000 pages (GPO).

NUREG/CP-0142, Vols. 1–4 *Proceedings of the 7th International Meeting on Nuclear Reactor Thermal–Hydraulics,* NURETH-7, September 10–15, 1995, Saratoga Springs, N.Y., R. C. Block and F. Feiner, American Nuclear Society, Ill., September 1995, 3200 pages (GPO).

NUREG/CR-1465 *Incentive Regulation of Investor-Owned Nuclear Power Plants by Public Utility Regulators,* M. D. McKinney, Pacific Northwest Labs., Wash., April 1995, 60 pages (GPO).

NUREG/CR-6004 *Probabilistic Pipe Fracture Evaluations for Leak-Rate-Detection Applications,* S. Rahman et al., Battelle, Ohio, April 1995, 310 pages (GPO).

NUREG/CR-6089 *Detection of Pump Degradation,* R. H. Greene and D. A. Casada, Oak Ridge National Lab., Tenn., August 1995, 93 pages (GPO).

NUREG/CR-6109 *The Probability of Containment Failure by Direct Containment Heating in Surry,* M. M. Pilch et al., Sandia National Labs., N.M., May 1995, 255 pages (GPO).

NUREG/CR-6112 *Impact of Reduced Dose Limits on NRC Licensed Activities. Major Issues in the Implementation of ICRP/NCRP Dose Limit Recommendations, Final Report,* C. B. Meinhold, Brookhaven National Lab., N.Y., May 1995, 64 pages (GPO).

NUREG/CR-6261 *A Summary of ORNL Fission Product Release Tests with Recommended Release Rates and Diffusion Coefficients,* R. A. Lorenz and M. F. Osborne, Oak Ridge National Lab., Tenn., July 1995, 70 pages (GPO).

NUREG/CR-6307 *Summary of Comments Received at Workshop on Use of a Site Specific Advisory Board (SSAB) to Facilitate Public Participation in Decommissioning Cases,* J. Caplin et al., June 1995, 95 pages (GPO).

NUREG/CR-6334 *New Sensor for Measurement of Low Air Flow Velocity. Phase I Final Report,* H. M. Hashemian et al., AMS Services Corp., Tenn., August 1995, 124 pages (GPO).

Other Items

ICRU Report 53 *Gamma-Ray Spectrometry in the Environment,* International Commission on Radiation Units and Measurements Inc. (ICRU), M.D., 1995, 85 pages (available from ICRU Publications, 7910 Woodmont Avenue, Suite 800, Bethesda, MD 20814-3095).

ORAU 95/F-30 *Final Report of the Committee on Interagency Radiation Research and Policy Coordination, 1984–1995,* Office of Science and Technology Policy, Washington, DC, September 1995, 90 pages (NTIS).

IAEA/PI/A22E *International Atomic Energy Agency (IAEA) 1995 Highlights of Activities,* IAEA, September 1995, 85 pages (available from UNIPUB, 4611-F Assembly Drive, Lanham, MD 20706-4391).

Nuclear Power Plant. Postgraduate Course on Energy Engineering, Distant Learning Package, M. Cumo and N. Afgan, University of Rome, 1995, 836 pages (available from Prof. Maurizio Cumo, Distant Learning Center, University of Rome, Viale Regina Margherita, 125 00198 Rome, Italy).

Aging and Life Extension of Major Light Water Reactor Components, Vikram N. Shah and Philip E. MacDonald (Eds.), Idaho National Engineering Laboratory, Idaho Falls, Idaho, 1993 (available from Elsevier Science Publishers, New York).

Atoms, Radiation, and Radiation Protection: Second Edition, James E. Turner, Oak Ridge National Laboratory, Martin Marietta Energy Systems, Inc., and adjunct professor, University of Tennessee, 1995 (available from John Wiley & Sons, Inc., New York).

Contemporary Health Physics: Problems and Solutions, Joseph John Bevelacqua, Wisconsin Electric Power Company, 1995 (available from John Wiley & Sons, Inc., New York).

REGULATORY GUIDES

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

Table 1 Regulatory Guides

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

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

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

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

Division 1 Power Reactor Guides

1.82 (Draft, Rev. 2) *Water Sources for Long-Term Recirculation Cooling Following Loss-of-Coolant Accident,* July 1995.

1.118 (Rev. 3) *Periodic Testing of Electric Power and Protection Systems,* April 1995.

1.149 (Draft, Rev. 2) *Nuclear Power Plant Simulation Facilities for Use in Operator License Exams,* June 1995.

1.152 (Draft, Rev. 1) *Criteria for Digital Computers in Safety System of Nuclear Power Plants,* May 1995.

1.161 *Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 ft-lb,* June 1995.

1.163 *Performance-Based Containment Leak-Test Program,* September 1995.

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 April 1995 through September 1995. 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 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 15.20-1994 (Published) *Decommissioning of Research Reactors,* $75.00.

BSR/ANS 8.7 [Revision of ANSI/ANS 8.7-1975(R1987) for comment] *Nuclear Criticality Safety in the Storage of Fissile Materials,* $15.00.

American Society for Testing and Materials

Standards prepared by ASTM can be obtained from ASTM, Attention: Customer Service Department, 1916 Race Street, Philadelphia, PA 19103.

BSR/ASTM E1539 (New standard, approved by ANSI/BSR) *Guide for Use of Radiation Indicators,* $15.00.

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 and ISO standards are included in this issue.

IEC

IEC 1772:1995 (Published) *Nuclear Power Plants—Main Control Room—Application of Visual Display Units (VDU),* $108.00.

ISO

ISO 12183:1995 (Published) *Controlled-Potential Coulometric Assay of Plutonium,* $36.00.

ISO/DIS 11933-2 (Draft) *Components for Containment Enclosures—Part 2: Gloves, Welded Bags, Remote Tong and Manipulator Gaiters,* $100.00.

Proposed Rule Changes as of June 30, 1995[a,b]

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

Number of part to be changed	Date published for comment	Date comment period expired	Date published; date effective	Topic or proposed effect	Current action and/or comment, *Federal Register* volumes and page numbers
10 CFR 1			1-20-95; 1-20-95	NRC Policy Statements; withdrawal	Policy statement withdrawn in 60:013 (4071)
10 CFR 2, 10 CFR 72	6-3-93	8-17-93; 10-1-93	4-28-95; 5-30-95	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); final rule in 60:082 (20879)
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	5-11-94	6-10-94		Summary reports on the status of petitions for rulemaking; frequency	Published for comment in 59:90 (24371)
10 CFR 2	8-23-94; 9-27-94; 11-28-94	10-24-94; 12-28-94		Reexamination of the NRC enforcement policy	Published for comment in 59:162 (43298); correction in 59:171 (46004); expanded scope in 59:186 (49215); revised in 59:227 (60697)
10 CFR 2, 51,54	9-9-94	12-8-94	5-8-95; 6-7-95	Nuclear power plant license renewal; proposed revisions	Published for comment in 59:174 (46574); final rule in 60:088 (22461)
10 CFR 2	11-30-94	12-30-94	4-11-95; 5-11-95	NRC size standards, proposed revision	Published for comment in 59:229 (61293); final rule in 60:069 (18344)
10 CFR 2	3-28-95	6-12-95		Petition for rulemaking; procedure for submission	Published for comment in 60:059 (15878)
10 CFR 2			6-30-95; 6-30-95	Policy and procedure for enforcement actions; removal	Final rule in 60:126 Part III (34380)
10 CFR 11 10 CFR 25	12-28-94	1-27-95	5-17-95; 6-16-95	NRC licensee renewal/reinvestigation program	Published for comment in 59:248 (66812); final rule in 60:095 (26355)
10 CFR 19 10 CFR 20	2-3-94	4-4-94		Radiation protection requirements; amended definitions and criteria	Published for comment in 59:023 (5132)
10 CFR 20	2-25-94	5-26-94		Disposal of radioactive material by release into sanitary sewer systems	Advanced notice of proposed rulemaking published in 59:038 (9146)

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Proposed Rule Changes as of June 30, 1995 (Continued)

Number of part to be changed	Date published for comment	Date comment period expired	Date published; date effective	Topic or proposed effect	Current action and/or comment, *Federal Register* volumes and page numbers
10 CFR 20 10 CFR 61	4-21-92	7-20-92	3-27-95; 3-1-98	Low-level waste shipment manifest information and reporting	Published for comment in 57:077 (14500); final rule in 60:058 (15649); correction to final rule in 60:094 (25983)
10 CFR 20	6-18-93	8-15-93; 9-20-93		Radiological criteria for decommissioning of NRC-licensed facilities; generic environmental impact statement (GEIS) for rulemaking, notice of intent to prepare a GEIS and to conduct a scoping process	Published for comment in 58:116 (33570); comment period extended in 58:154(42882)
10 CFR 20	2-2-94	3-11-94		Radiological criteria for decommissioning of NRC-licensed facilities; enhanced participatory rulemaking, availability of the staff's draft of the rule	Published for comment in 59:022 (4868)
10 CFR 20 10 CFR 35	6-15-94	8-29-94		Criteria for the release of patients administered radioactive material	Published for comment in 59:114 (30724)
10 CFR 20, 30,40,50, 51,70,72	8-22-94	12-20-94; 1-20-95		Radiological criteria for decommissioning	Published for comment in 59:161, Part III (43200); comment period extended in 59:236 (63733)
10 CFR 20, 30,40,61, 70,72	12-28-94	3-28-95		Termination or transfer of licensed activities: recordkeeping requirements	Published for comment in 59:248 (66814)
10 CFR 20 10 CFR 35	1-25-95	4-10-95		Medical administration of radiation and radioactive materials	Published for comment in 60:016 (4872)
10 CFR 20			2-10-95; 3-13-95	Frequency of medical examinations for use of respiratory protection equipment	Final rule in 60:028 (7900)
10 CFR 20			4-25-95; 4-25-95	Standards for protection against radiation; clarification	Final rule in 60:079 (20183)
10 CFR 21	10-24-94	1-9-95		Procurement of commercial grade items by nuclear power plant licensees	Published for comment in 59:204 (53372)

Proposed Rule Changes as of June 30, 1995 (Continued)

Number of part to be changed	Date published for comment	Date comment period expired	Date published; date effective	Topic or proposed effect	Current action and/or comment, *Federal Register* volumes and page numbers
10 CFR 26	5-11-94	9-9-94		Consideration of changes to fitness-for-duty (FFD) requirements	Published for comment in 59:090 (24373)
10 CFR 30, 32,35	6-17-93	10-15-93	12-2-94; 12-19-94; 1-1-95	Preparation, transfer for commercial distribution, and use of byproduct material for medical use	Published for comment in 58:115 (33396); final rule in 59:231 (61767); correction to final rule in 59:242 (65243); clarification to final rule in 60:002 (322)
10 CFR 30, 40,70,72	6-22-94	9-20-94		Clarification of decommissioning funding requirements	Published for comment in 59:119 (32138)
10 CFR 32			1-19-95; 12-31-94	Requirement to report transfers of devices to generally licensed persons	Final rule in 60:012 (3735)
10 CFR 34 10 CFR 150	2-28-94	5-31-94		Licenses for radiography and radiation safety requirements for radiographic operations	Published for comment in 59:039 (9429)
10 CFR 34			5-31-95; 6-30-95	Performance requirements for radiography equipment	Final rule in 60:104 (28323)
10 CFR 35	11-3-94	3-3-95		Request for comments regarding potential modifications of NRC's therapy regulations	Published for comment in 59:212 (55068)
10 CFR 50 10 CFR 52 10 CFR 100	10-20-92	2-17-93; 3-24-93; 6-1-93; 2-14-95; 5-12-95		Reactor site criteria, including seismic and earthquake engineering criteria for nuclear power plants and proposed denial of petition for rulemaking from Free Environment, Inc., et al.	Published for comment in 57:203 (47802); comment period extended in 58:002(271); extended again in 58:057 (16377); extended again in 59:199 (52255); extended again in 60:026 (7467); extension deadline set 60:039 (10810)
10 CFR 50	6-28-93; 4-14-95	9-13-93; 7-13-95		Production and utilization facilities; emergency planning and preparedness-exercise requirements	Published for comment in 58:122 (34539); published for comment in 60:072 (19002)
10 CFR 50	1-7-94	3-24-94; 4-25-94		Codes and standards for nuclear power plants; subsection IWE and subsection IWL	Published for comment in 59:005 (979); comment period extended in 59:059 (4373)

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Proposed Rule Changes as of June 30, 1995 (Continued)

Number of part to be changed	Date published for comment	Date comment period expired	Date published; date effective	Topic or proposed effect	Current action and/or comment, *Federal Register* volumes and page numbers
10 CFR 50	9-19-94	12-5-94		Steam generator tube integrity for operating nuclear power plants	Published for comment in 59:180 (47817)
10 CFR 50	9-20-94	12-5-94		Technical specifications	Published for comment in 59:181 (48180)
10 CFR 50	10-4-94	1-3-95		Fracture toughness requirements for light water reactor pressure vessels	Published for comment in 59:191 (50513)
10 CFR 50	10-19-94; 10-25-94; 1-18-95	1-3-95; 2-3-95		Shutdown and low-power operations for nuclear power reactors	Published for comment in 59:201 (52707); correction in 59:205 (53613); comment period extended in 60:011 (3579)
10 CFR 50, 55,73	11-2-94	12-19-94	3-14-95; 4-13-95	Reduction of reporting requirements imposed on NRC licensees	Published for comment in 59:211 (54843); final rule in 60:049 (13615)
10 CFR 50	2-21-95	5-8-95		Primary reactor containment leakage testing for water-cooled power reactors	Published for comment in 60:034 (9634)
10 CFR 50 10 CFR 70	4-17-95	5-17-95		Physical security plan format changes	Published for comment in 60:073 (19170)
10 CFR 51	9-17-91	12-16-91; 3-16-92; 9-8-94		Environmental review for renewal of operating licenses	Published for comment in 56:180 (47016); comment period extended in 56:228 (59898); supplemental proposed rulemaking in 59:141 (37724)
10 CFR 52	11-3-93	1-3-94		Rulemakings to grant standard design certification for evolutionary light water reactor designs	Advance notice of proposed rulemaking published in 58:211 (58664)
10 CFR 52			1-22-93	Combined Licenses; Conforming Amendments; Post-Promulgation Comment	Notice of post-promulgation comment in 60:016 (4877)
10 CFR 52	4-7-95	8-7-95		Standard design certification for the U.S. Advanced Boiling Water Reactor design	Published for comment in 60:067 (17902)
10 CFR 52	4-7-95	8-7-95		Standard design certification for the System 80+ design	Published for comment in 60:067 (17924)

Proposed Rule Changes as of June 30, 1995 (Continued)

Number of part to be changed	Date published for comment	Date comment period expired	Date published; date effective	Topic or proposed effect	Current action and/or comment, *Federal Register* volumes and page numbers
10 CFR 55	5-20-93	7-19-93		Operator's licenses	Published for comment in 58:096 (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 60	**3-22-95**	**6-20-95**		**Disposal of high-level radioactive wastes in geologic repositories; design basis events**	**Published for comment in 60:055 (15180)**
10 CFR 61	8-3-94	10-3-94; 12-2-94		Land ownership requirements for low-level waste sites	Published for comment in 59:148 (39485); comment period extended in 59:202 (52941)
10 CFR 72	5-24-93	8-9-93; 11-9-93	**6-22-95; 9-20-95**	Emergency planning licensing requirements for independent spent fuel facilities (ISFSI) and monitored retrievable storage facilities (MRS)	Published for comment in 58:098 (29795); comment period extended in 55:166 (45463); **final rule in 60:120 (32430)**
10 CFR 73	**5-10-95**	**6-9-95**		**Changes to nuclear power plant security requirements associated with containment access control**	**Published for comment in 60:090 (24803)**
10 CFR 170 10 CFR 171	4-19-93	7-19-93		NRC fee policy; request for public comment	Published for comment in 58:073 (21116)
10 CFR 170 10 CFR 171	**3-20-95**	**4-19-95**	**6-20-95; 7-20-95**	**Revision of fee schedules; 100% fee recovery, FY 1995**	**Published for comment in 60:053 (14675); correction in 60:062 (16589); correction in 60:071 (18882); final rule in 60:118 (32218); corrections in 60:124 (33462)**

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

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

The Authors

The Chornobyl Accident Revisited, Part III: Chornobyl Source Term Release Dynamics and Reconstruction of Events During the Active Phase

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Second ANS Workshop on the Safety of Soviet-Designed Nuclear Power Plants

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Elements of a Nuclear Criticality Safety Program

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Rickover, Excellence, and Criticality Safety Programs

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Transient Analysis of the PIUS Advanced Reactor Design with the TRAC-PF1/MOD2 Code

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The Hierarchy-By-Interval Approach to Identifying Important Models that Need Improvement in Severe-Accident Simulation Codes

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RELAP5/MOD3 Code Coupling Model

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Missiles Caused by Severe Pressurized-Water Reactor Accidents

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Validation of COMMIX with Westinghouse AP-600 PCCS Test Data

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Spent Nuclear Fuel Characterization for a Bounding Reference Assembly for the Receiving Basin for Off-Site Fuel

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Reactor Coolant System Blowdown at Wolf Creek on September 17, 1994

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Reviewers of *Nuclear Safety,* Volume 36

The technical quality of a journal depends not only on the competence and efforts of its authors and editorial staff but also, to a major extent, on the dedication of its corps of peer reviewers. We wish to acknowledge gratefully the many technical experts whose voluntary and unrewarded reviews of proposed *Nuclear Safety* articles have been indispensable in the selection of articles and in the revision of articles to prepare them for publication.

We list below all the names of those who reviewed articles for publication in Vol. 36, whether the articles were used or not. Since it is our policy not to reveal the reviewers' identities to the authors, all reviewers are listed in alphabetical order together with their affiliations.

This list does not include, though we are most grateful to them also, the names of the NRC staff members who review all *Nuclear Safety* articles to ensure that the policies and positions of their agency are not misstated or distorted.

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RADIATION BIOLOGY AND RADIATION PROTECTION—MODERN DEVELOPMENTS AND TENDENCIES IN RADIATION BIOLOGY

City of Hannover, Germany, October 23–25, 1996

Research in radiation biology is receiving increasing importance in the field of radiation protection. The topics in this conference cover all kinds of activities in research and work on biological effects of ionizing radiation. The following program structure shows the range of topics to be treated:

- **A. Mechanisms of Radiation Effects**
  1. Molecular and cellular mechanisms of radiation effects
  2. Genetic and prenatal radiation effects (experimental/epidemiological)
  3. Cancerogenesis/Mutagenesis
- **B. New Methods in Research on Radiation Biology**
  1. Micro dosimetry under biological aspects
  2. New molecular biological methods, including computer simulation of DNA
  3. Procedures of biological dosimetry
  4. Neutrons and dense ionizing radiation (plutonium, radon) in radiation biology
- **C. Results and Concepts for Radiation Protection**
  1. Individual radiation sensitivity
  2. Radiation biological aspects in medical application of ionizing radiation
  3. Industrial accidents
  4. Radiation exposition due to air transportation and space missions

The Conference President/Conference Secretariat is Dr. Günter Heinemann, c/o Kernkraftwerk Stade, P.O. Box, D-21683, Stade, Germany, Tel: +49-4141-772955, Fax: +49-4141-799454.

1997 IEEE SIXTH CONFERENCE ON HUMAN FACTORS AND POWER PLANTS

Orlando, Florida, June 7–12, 1997

The Institute of Electrical and Electronic Engineers (IEEE) Power Engineering Society's Nuclear Power Engineering Committee announces a sixth conference on the role of Human Factors in Nuclear Power Plants. The theme of this conference will be Human Factors and Power Generation: A Global Perspective.

For general conference information, contact Dr. Harold Blackman, General Chairman, 1997 IEEE Sixth Conference on Human Factors and Power Industry, c/o Lockheed Martin Idaho Technologies, P.O. Box 1625, Idaho Falls, ID 83415, Fax: (208) 526-0425, E-mail: HSB@INEL.GOV

To be placed on the mailing list for conference announcements, contact Mr. Stephen Fleger, Publicity Coordinator, 1997 IEEE Sixth Conference on Human Factors and Power Industry, c/o SAIC, 11251 Roger Bacon Drive, Reston, VA 22090, Fax: (703) 709-1038, E-mail: Stephen.A.Fleger@cpmx.saic.com

For individuals wishing to submit a technical paper for consideration, contact Dr. Jay Persensky, Technical Chairman, U.S. Nuclear Regulatory Commission, T 10 E 33, Washington, DC 20555, Fax: (301) 415-5160, E-mail: JJP2@NRC.GOV