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**SOURCE-TERM REEVALUATION FOR U.S. COMMERCIAL
NUCLEAR POWER REACTORS: A STATUS REPORT**

by

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PREFACE

This document reviews existing source-term information and reports on the efforts of the U.S. Nuclear Regulatory Commission (NRC) Source Term Task Force and others in developing new source-term methodology and data. The report results from a project conducted by Argonne National Laboratory for the U.S. Department of Energy (DOE) Office of Nuclear Safety. This particular report, which summarizes the current status of NRC and industry source-term reassessment for commercial light-water nuclear power reactors, represents one aspect of the work being done by Argonne National Laboratory on the application of source-term technology to DOE reactors. A review of phenomenological considerations in source-term analysis as they pertain to typical LWR reactors is in progress and will be reported on in the future.

This document summarizes a broad-scoped, rapidly moving field. It reflects the status of source-term reevaluation for U.S. commercial nuclear power reactors as of September 30, 1984, and is based on information presented publicly, published in open literature, or published in unclassified limited-distribution reports. Periodic updates will be issued when warranted on the basis of major developments in the field.

The text of this report contains not only direct quotations but also numerous references to the opinions and judgments of many different organizations and individuals involved in source term reassessment. These positions do not necessarily represent those of DOE, Argonne National Laboratory, or Argonne and DOE technical staff members, except where explicitly indicated.

Only the reevaluation of source terms for U.S. commercial light-water reactors is reviewed in this report, and the results are not directly applicable to other reactor types. However, this information is expected to provide guidance to studies of specific reactor facilities that are currently in progress within DOE.

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SOURCE-TERM REEVALUATION FOR U.S. COMMERCIAL NUCLEAR POWER REACTORS: A STATUS REPORT

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SUMMARY

This report, which was prepared for the U.S. Department of Energy's Office of Nuclear Reactor and Facility Safety, summarizes progress in source-term reevaluation for U.S. commercial nuclear power reactors. Only results that had been discussed publicly, had been published in the open literature, or were available in preliminary reports as of September 30, 1984, are included here. More than 20 organizations are participating in source-term programs, which have been undertaken to (1) examine severe accident phenomena in light-water power reactors (including the chemical and physical behavior of fission products under accident conditions), (2) update and reevaluate source terms, and (3) resolve differences between predictions and observations of radiation releases and related phenomena. Results from these source-term activities have been documented in over 100 publications to date.

SIGNIFICANT ACTIVITIES

Source-term reassessment for commercial light-water power reactors is currently the subject of intense activity by the U.S. Nuclear Regulatory Commission (NRC). The primary contribution to the NRC source-term reassessment comes from studies conducted at Battelle Columbus Laboratories for five power plants: Surry 1 and 2, pressurized-water reactors in Gravel Neck, Virginia; Zion 1 and 2, pressurized-water reactors in Zion, Illinois; Peach Bottom 2 and 3, boiling-water reactors in Peach Bottom, Pennsylvania; Grand Gulf 1 and 2, boiling-water reactors in Gibson, Mississippi; and Sequoyah 1 and 2, pressurized-water reactors in Daisy, Tennessee.

The NRC staff will assess the risk and regulatory significance of reassessed source terms on the basis of analyses and findings of NRC contractors (including Battelle, Oak Ridge National Laboratory, and Sandia National Laboratories), the results of technical peer reviews of individual programs, work by the Industry Degraded Core Rulemaking (IDCOR) working group, and the resolution of certain issues by special task groups of the NRC staff. The final source-term reassessment will culminate in an NRC report, NUREG-0956. This final document was scheduled for release in December 1984 but is expected to be delayed until 1985. Any decision on lowering official source terms or altering any regulations based on them will probably await the NRC's conclusions from ongoing research and the American Physical Society's peer review of research methodology, which is not likely to be released before February 1985. Thus, the NRC source-term report and staff recommendations for reductions in source terms are not expected to be issued before March 1985.

A parallel effort to the NRC program has been undertaken by the IDCOR working group under the aegis of the Atomic Industrial Forum. The IDCOR project is expected to be completed in December 1984.

Other participants in source-term investigations include the American Nuclear Society, the American Physical Society, Stone and Webster Engineering Corporation, and Electric Power Research Institute. Their work is in progress.

Many countries are undertaking a wide range of experimental programs to obtain better data on fission product release after severe accidents. These programs range in size from relatively small laboratory-scale experiments to determine the physico-chemical characteristics of fission product releases to full-size mock-up experiments on containments, components, and reactor loops. Several of these projects are internationally sponsored.

RESULTS

Redefinitions of some source terms are being proposed on the basis of significant advances in source-term assessment technology that have taken place since the NRC's *Reactor Safety Study — An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants* (WASH-1400).¹ In many important accident sequences, the radionuclide releases to the atmosphere predicted with current methodology are substantially lower than those predicted in WASH-1400. As a typical example, source terms based on current methodology (as exemplified by recent work at Battelle Columbus Laboratories) and WASH-1400 results are shown in Table S.1 for an accident sequence at a Surry pressurized-water reactor.

The most significant noble-gas fission products are isotopes of krypton and xenon. Most past and present studies assume that, if an escape pathway opens, all of the noble gases in the core inventory will be released.

Battelle's values for iodine (5% of the core inventory), which are based on current methodology, are significantly lower than the values in WASH-1400 (70%). The lower values for radiiodine stem from current thinking that, under light-water reactor accident conditions (i.e., a reducing environment), iodine would be released from the core largely as cesium iodide (rather than as iodine vapor) and the cesium iodide would be essentially all dissolved in water.

Battelle's values for cesium and rubidium (4%) are significantly lower than the values in WASH-1400 (40%). Cesium and rubidium would be released as vapor if the fuel reached the melting point. However, they might subsequently condense to form aerosols. A fraction of the cesium would be expected to combine with iodine to form water-soluble cesium iodide. Most of the cesium would combine with water to form cesium hydroxide, which is highly soluble in water. Cesium hydroxide can also react with stainless steel and other materials to form stable compounds, thus further reducing the amount of cesium released.

TABLE S.1 Source-Term Comparison for a TMLB'-6 Accident at a Surry Pressurized-Water Reactor^a

Study	Study Date	Percentage of Core Inventory Released to Atmosphere						
		Xenon-Krypton	Iodine	Cesium-Rubidium	Tellurium-Antimony ^b	Barium-Strontium	Ruthenium ^c	Lanthanum ^d
WASH-1400	1975	100	70	40	20	5	2	0.3
Battelle	1984	100	5	4	10	5	0.5	0.1

^aTMLB'-6 Sequence: Transient event; failure of secondary-system steam relief valves and the power conversion system; failure of secondary-system steam relief valves and the auxiliary feedwater system; failure to recover either on-site or off-site electric power within about 1 to 3 hours of an initiating transient event that is a loss of off-site AC power; containment failure due to overpressure.

^bTellurium includes selenium.

^cRuthenium includes molybdenum, palladium, rhodium, and technetium.

^dLanthanum includes neodymium, europium, yttrium, cerium, praseodymium, promethium, samarium, neptunium, plutonium, zirconium, and niobium.

Source: Refs. 1 and 2.

Battelle's values for the remaining fission products are lower than or the same as those in WASH-1400. Tellurium and antimony, barium and strontium, ruthenium, and lanthanum would be expected to emerge as aerosols with increasing severity of core damage. Tellurium and ruthenium might be scavenged by molten steel and zirconium, thus reducing their availability.

Studies since WASH-1400 have shown that various aerosol removal processes, including condensation, gravitational settling, diffusiophoresis, and thermophoresis, have a significant effect on retention of fission products within plant components and structures. In particular, the suspended liquid (water droplets) associated with the two-phase nature of the post-accident atmosphere has been shown to enhance the agglomeration and settling of aerosols within the plant components and structures.

Single-control-volume analyses of complex structures, such as containments for pressurized-water reactors, substantially overestimate releases. Multicompartment analyses of these structures result in lower source terms.

Containments have been shown to be much stronger than assumed in WASH-1400. They require longer times to breach, which in turn allow a greater degree of aerosol depletion and natural decay. There appears to be a substantial consensus that steam explosions and hydrogen deflagration will not lead to early containment failure. The development of distributed containment breaching (cracking) may allow small quantities of radioactivity to escape.

Radioactivity released from reactor containments, particularly by way of pre-existing penetrations, generally escapes into other structures outside the containment

rather than directly to the atmosphere. Since WASH-1400 was published, retention of fission products in these structures has been shown to be substantial.

Uncertainties exist in analyses of basic thermal hydraulic behavior; such uncertainties include changes in core geometry as a melt progresses. Additional uncertainty exists about resuspension and revolatilization of deposited material and recirculation flows, and more knowledge is needed about core-concrete interactions and fission product reactions. However, these uncertainties are not expected to cause major changes in source terms based on current methodology.

IMPORTANT CONCLUSIONS

The important general conclusions of various source-term programs sponsored by government and industry can be summarized as follows:

- There is consensus that WASH-1400 source terms and the assumptions that form the regulatory basis for siting and emergency response planning are more conservative than originally recognized.
- A preponderance of evidence indicates that source terms for most severe accident sequences can be reduced by an order of magnitude or more (noble gases excepted). Source terms for a few serious accident sequences (outliers) remain large. This issue is still under investigation and is still being evaluated by the NRC.
- Containment failure, if occurring at all, may be expected to occur later rather than earlier and will likely be a distributed breaching (cracking) rather than gross failure. This type of failure would allow small radioactivity leakage rather than a large burst-type release. However, consensus has not been reached that early catastrophic containment failures from steam or hydrogen explosions are impossible.
- Current scientific and technical understanding is sufficient to form a basis for redefinition of source terms. The detailed phenomenology of severe accidents is still not fully understood and is the subject of continuing investigations. However, it is expected that a better understanding of these phenomena will probably not have a major effect on current source-term assessments.
- Improving the source-term values is most important for emergency response planning. Should NRC adopt appreciably reduced source terms, emergency planning regulations might be modified to reduce the areas of evacuation zones or to adopt a policy of sheltering rather than evacuation. However, the NRC may retain the 10-mi-radius emergency planning zones that are now in place, possibly supplementing them with a graded response based on subzones

requiring different degrees of readiness and different types of protective action.

- Work on source-term reevaluation is being conducted with similar methods worldwide, although practical results may differ for legal or other reasons.

REFERENCES FOR SUMMARY

1. *Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants*, U.S. Nuclear Regulatory Commission Report NUREG-75/014 (previously published as WASH-1400) (Oct. 1975).
2. Gieseke, J.A., et al., *Radionuclide Release under Specific LWR Accident Conditions: Volume V, PWR, Large, Dry Containment Design (Surry Plant Recalculations)*, Battelle Columbus Laboratories Draft Report BMI-2104, Vol. V (July 1984).

1 INTRODUCTION

Radionuclide release from severe accidents at nuclear reactors has been a subject of continuing significance since the earliest days of reactor development, not only for commercial nuclear power plants but also for other large nuclear reactors.¹⁻¹¹ The accurate assessment of source terms (the amount and type of radionuclides that could escape to the environment from a nuclear reactor in a specific accident sequence) for commercial light-water power reactors has recently become a high-priority issue within the utility industry and government.^{2,12-15} Both the U.S. Nuclear Regulatory Commission (NRC) and the nuclear power industry are actively addressing this question; industry-sponsored research programs, federally funded research-and-development programs, independent scientific studies, and international activities are all in progress.^{12-14,16,17}

This report summarizes these current activities. It reviews progress (as of September 30, 1984) in the broad, rapidly changing field of source-term reevaluation. Ultimately, results of the source-term reevaluation will be used to improve emergency planning procedures and the mitigation of accident effects.

2 BACKGROUND

The principal direct risk to the public from nuclear reactors arises from the intensely radioactive nuclei (principally fission products) generated as nuclear reactions take place within the reactor. In normal operations, these fission products reside almost entirely within the nuclear fuel where they are created. However, under accident conditions some of these fission products can be released from the fuel and escape to the biosphere.¹⁸⁻²¹ The amount and type of radioactivity escaping is referred to as the source term. In reactor safety analysis, off-site consequences are evaluated on the basis of the estimated release of radioactive materials to the environment and the consequent exposure of people and property.^{4,22-24}

To estimate the potential risk to the public from nuclear reactor accidents, one must first establish the probabilities of occurrence of the various ways or accident sequences in which the fission products can be released from the nuclear fuel to the biosphere. Event-tree/fault-tree analysis can be used to establish such probabilities. Event trees define the hypothetical accident sequences, and fault trees permit estimation of their likelihood of occurrence. The physical and chemical phenomena involved in these accident sequences must be analyzed to estimate the source term for each accident sequence. Finally, one must predict how the radioactivity is dispersed in the biosphere so that population exposure, contamination levels, and health effects from the postulated accident can be evaluated.

The first full analysis of this type was documented in *Reactor Safety Study — An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants* (WASH-1400), which was initiated by the U.S. Atomic Energy Commission and completed in 1975 under the sponsorship of the U.S. Nuclear Regulatory Commission.¹ Generally, this and other studies of reactor safety predicted very small risks to the public from nuclear power plant accidents in most cases; however, the predicted consequences of certain low-probability, high-risk accident sequences were severe.

The source-term issue was raised again in the fall of 1979.²⁵ Current discussion centers on whether the greatest predicted consequences of reactor accidents have been overestimated in past studies because the source terms were too conservatively evaluated and thus too large.¹²⁻¹⁴ A more realistic approach to the severe-consequence accidents is being sought. Recent information and reevaluations of light-water reactor accident phenomena have suggested that source-term assumptions in current regulatory use may be overly conservative and that modifications to the source term may be appropriate.²⁶⁻³¹

The amount of radioactivity released to the environment from the accident at the Three Mile Island 2 nuclear power plant for all radionuclides except the noble gases was substantially less than predicted.^{12,27-29} This observation (which confirmed some prior suppositions) led industry representatives to seek a reevaluation of source terms for accidents involving severe core damage.^{26,32} As a consequence, government and nuclear power industries in the United States and abroad have undertaken substantial efforts to (1) examine severe-accident phenomena, including the chemical and physical accident behavior of fission products, (2) update and reevaluate source terms, and (3) resolve differences between source term predictions and observations.

3 REASSESSMENT OF SOURCE TERMS

3.1 INTRODUCTION

Table 1 compares some of the various historical and current severe-accident source terms that have been in use or proposed. Clearly, assumptions and conclusions about the percentages of core inventory released to the biosphere after severe reactor accidents have varied widely. Severe accidents are those in which fuel overheats, substantial amounts of fission products escape to the containment, and the containment fails to prevent the escape of significant amounts of radioactivity to the environment.

The release to the biosphere of fission products from a severely damaged core is generally strongly affected by two major factors: the timing of the accident sequence creating the pathway for escape and the physical and chemical nature of the fission products.

The timing of the accident sequence creating the pathway for escape is itself a major determinant of the overall magnitude of the fission product release. It is also a determinant of the relative amounts of short-lived versus longer-lived fission products released. The timing of the release depends strongly upon considerations such as the containment failure mode. (The most-serious accidents with the largest overall releases involve early breach of containment or containment bypass.)

The relative abundance in the release of the chemically different fission products depends both on the characteristics of the accident (for example, the maximum temperature to which the fuel is heated) and the physical and chemical properties of the particular radionuclide. These properties also significantly affect retention of the radionuclides throughout the transport process -- during their release from the fuel, transport in the primary coolant system to the containment, and transport through the containment.⁵ Evaluating the effect of these physical and chemical processes on radionuclide retention at each stage of the accident sequence is one of the main ways in which source-term estimates are being reduced.

The analysis of a nuclear accident may involve hundreds of different fission products. Thus, in examinations of fission product behavior, it is helpful to group together isotopes of the same element, and more generally to group together elements that behave similarly. Since release of fission products from heated fuel (as well as in other phenomena) depends on the volatilities of the elemental fission products, fission product groups are conventionally arranged in order of decreasing volatility, as in the categories listed in Table 1 and discussed further in the next section.

3.2 RADIONUCLIDES

During and after reactor operation, many fission product species exist in the reactor fuel. The potential consequences of radionuclides escaping into the environment in a severe reactor accident vary in importance. The factors determining this importance of a radionuclide are (1) its total inventory in the reactor, (2) the physical

TABLE 1 Comparison of Source Terms for Various Severe Accidents

Organization and Report No. (if available)	Accident Sequences ^a	Study Date	Percentage of Core Inventory Released							Release to Containment (C) or Environment (E)
			Xenon- Krypton	Iodine	Cesium- Rubidium	Tellurium- Antimony	Barium- Strontium	Ruthenium	Lanthanum	
Atomic Energy Commission, TID-14844 ^{b,c}	Not applicable	1962	100	25	1	1	1	1	1	C
NRC, WASH-1400 ^{b,c,d}	Release Category PWR-2	1975	90	70	50	30	6	2	0.4	C,E
German Risk Study ^{b,c}	Release Category 2	1979	100	39	26	16	3	-	-	C
NRC, NUREG-0771 ^e	Group 1 "Worst Case"	1981	100	30-70	30-70	30-70	1-10	1-40	0.1-0.5	E
German Project for Nuclear Safety ^{b,c,d}	Release Category 2	1982	100	0.64	0.69	0.56	0.69	-	-	C,E
Sandia, SAND 81-1549 ^{d,f}	SST-1	1982	100	45	67	64	7	5	0.9	E
Stone and Webster ^{b,c,d}	Not applicable	1982/3	100	1	1	1	1	1	0.4	C,E
Indian Point Testimony ^d	Not applicable	1983	100	1.5	1.0	0.1	0.1	0.1	0.1	E
Battelle, NUREG-0956 ^d	TMLB-γ sequence	1983	100	70	60	50	1	0.08	0.2	Z
Battelle, BMI-2104 ^g	PWR-2, TMLB'-δ	1983	100	70	60	50	1	0.08	0.2	E
Battelle, BMI-2104 ^h	PWR-2, TMLB'-δ ^e	1984	100	7	6	10	6	0.5	0.1	E
Stone and Webster ⁱ	Not applicable	1984	100	1	1	1	0.4	0.3	0.02	E

^aThe TMLB accident sequences are defined in App. B.
Other accident sequences are defined within each
study for the purpose of that study.

^bAdapted from Ref. 28.

^cAdapted from Nuclear Power Engineering, Atomics Summary.

^dAdapted from Ref. 29.

^eAdapted from Ref. 2.

^fAdapted from Ref. 33.

^gAdapted from Ref. 34.

^hAdapted from Ref. 35.

ⁱAdapted from Ref. 36.

and chemical properties determining its behavior in the plant and the environment, and (3) its biological characteristics. Some of these factors are inherent and others depend on features of the accident and plant design; thus, the importance of a radionuclide may depend to a significant extent on specific aspects of the particular accident sequence being considered (see Sec. 3.5).

The noble gases, though very volatile, are chemically inert and thus have a reduced importance in severe accidents because this inertness leads to low radiological consequences. The major noble-gas fission products are isotopes of krypton and xenon. Because they are chemically inert, the noble gases would not be expected to undergo chemical and physical interactions that might reduce the quantities released. Therefore most studies assume now, as previously, that all or almost all noble gases in the core inventory will eventually escape if an escape pathway opens. Table 1 reflects this assumption; all studies listed show 100% release of the core inventory of noble gases to the biosphere, except the WASH-1400, which shows a 90% release. Thus, substantial agreement exists on the noble gas component of source terms.^{14,29}

In terms of radiological consequences, radioiodine has been considered one of the most important components of the source term because of its relatively high abundance, high biological activity (iodine concentrates in the thyroid), and previously assumed form as an elemental gas that can be readily transported. Earlier severe-accident studies and regulations based on them thus focused on radioiodine as the principal substance of concern. However, more-recent evidence suggests that far less radioiodine than previously thought will escape during severe reactor accidents. The radioiodine component of the source term has thus become the subject of great interest and controversy. As Table 1 shows, values for iodine advocated in various studies differ by a factor of more than 100, ranging from a 70% release (WASH-1400 release category PWR-2 and Battelle accident sequence PWR-TMLB') to a 0.64% release (German Project for Nuclear Safety release category 2). (Note that these values are not for identical assumptions and accident sequences.)

Major considerations that may lead to reducing the radioiodine component of the source term are as follows: (1) under light-water reactor accident conditions (reducing environment), iodine would be released from the core and transported largely as cesium iodide, and (2) the cesium iodide would be dissolved in water. However, the net effect of these factors on the radioiodine release is still under discussion.^{2,12,14,28,29,37}

Other factors affecting the release of iodine include the time scale of the accident and the role of different chemical forms of iodine, including volatile molecular iodine, water-soluble inert cesium iodide, organic forms such as methyl iodide, and reaction products with water, such as iodine hydroxide.^{38,39} For example, while cesium hydroxide can react with stainless steel and boron carbide (in control rods) to form stable compounds, reducing the cesium component of the source term, cesium-iodine reactions with those surfaces produce free iodine, which is volatile and could escape easily.⁴⁰

Alkali metals -- cesium and rubidium -- make up another important category of radionuclides. However, as for iodine, the importance of cesium in the source term appears to have been overestimated. Alkali metal elements would be released as vapor if the fuel reached the melting point, then condense to form aerosols.¹⁴ However, it now

appears that most of the cesium in severe light-water reactor accidents would be released and transported predominantly as cesium hydroxide, because cesium combines with water to form cesium hydroxide, which is highly soluble in water. Cesium hydroxide can react with stainless steel and boron carbide (in control rods) to form stable compounds, thus also reducing the amount of cesium released.⁴⁰ A fraction of the cesium would combine with iodine to form water-soluble cesium iodide. Like iodine, alkali metals would be released before other aerosols that would be produced during a core melt.⁴¹ As Table 1 shows, values for the alkali metals differ by over a factor of 100, ranging from a high of about 70% of core inventory to a low of 0.69%.

Tellurium and antimony, barium and strontium, ruthenium, and lanthanum would be expected to emerge as aerosols with increasing severity of core damage.¹⁴ Tellurium and ruthenium might be scavenged by the molten steel and zirconium in the vicinity.¹⁴ As Table 1 shows, source-term values for these groups have ranged over two orders of magnitude or more, except for the lanthanum group, for which values have ranged over one order of magnitude, from 0.1% to 1%. Controversy continues over the behavior of the high-activity fission product tellurium, which is released late in a core melt; it can react either with stainless steel and Inconel, and plate out harmlessly, or with silver or tin from control rods, and become an aerosol available for release.⁴⁰

3.3 CHEMICAL AND PHYSICAL PROCESSES

A number of chemical and physical processes play important roles in severe accidents. These processes are involved in the escape of fission products from overheated fuel and the transport and behavior of fission products in the reactor coolant system, containment structure, and contiguous structures. Brief summaries of some of these processes follow.

3.3.1 Chemical Forms and Chemical Interactions

The fission products released from severely damaged fuel are subsequently transported through the reactor coolant system, usually either in elemental form or as a corresponding oxide; the important exceptions are halogens (iodine, bromine), alkali metals (cesium, rubidium), chalcogens (tellurium, selenium), and alkaline earths (strontium, barium).²³

The predominant form of fission product iodine is expected to be cesium iodide, because of the reducing environment. Tellurium, which is an iodine precursor in the radioactive decay of fission products, behaves differently chemically during the course of an accident; thus smaller amounts of radioactive iodine may form in regions devoid of cesium iodide. This radioiodine addition to the source term is expected to be minor. A hydrogen burn or radiation effects may increase the production of volatile organic iodides in the containment building, but the concentrations produced are expected to be small.²³

Because about 10 times more cesium is present than iodine, and because water is available, the predominant form of fission product cesium in the reactor coolant system

is generally expected to be cesium hydroxide. Cesium hydroxide can react with metal oxide surfaces or with boron control rod or reactor shim materials. However, the dominant processes are expected to be the dissolution of these highly soluble chemical forms in water and their formation of aerosols.²³ Additional phenomena associated with the chemical forms and interactions of fission products are discussed in Secs. 3.2 and 5.

3.3.2 Aerosol Effects

In a severe reactor accident, energy from the decay of fission products, exothermic reaction of overheated fuel cladding with steam, and, in some accidents, continued fission can cause the fuel and core material to heat up and vaporize more-volatile fission products, components of control rods, and structural material. These vapors can (1) nucleate to form aerosols or (2) condense on cooler surfaces of the reactor coolant system or on other aerosol particles. The physical and chemical aspects of aerosol formation and behavior must be analyzed to determine the amount of these materials escaping to containment or contiguous buildings.²³

3.3.3 Escape of Materials from a Degraded Core

Fission products and other materials escape from the degraded reactor core during fuel overheating, fragmentation and liquefaction of fuel rods, and formation of a debris bed and reaction between the core debris and the concrete base mat. A large fraction of the volatile fission products is expected to escape before fuel degradation.²³

As the core heats up, non-fission-product materials escape in far larger quantities than do fission products. These releases are important because the resultant aerosols strongly influence the transport and attenuation of airborne fission products in the reactor coolant system and the containment.²³

If the hot core debris penetrates the reactor vessel and contacts the base mat, the concrete could be eroded by the debris, possibly generating combustible gases (hydrogen and carbon monoxide) and substantial quantities of aerosols. These aerosols would consist mostly of nonradioactive material, but would carry some quantities of fission products. Fission products could escape principally by sparging of the gases created during concrete erosion.²³

3.3.4 Transport and Retention Processes in the Reactor Coolant System and Containment

Fission product retention in the reactor coolant system was not taken into account in WASH-1400 because of uncertainties and the need for conservative evaluation. More-recent analyses suggest that a significant fraction of the vaporized fission products would be retained in upper internal structures of the reactor vessel and, in some accidents, other parts of the reactor coolant system. The deposited materials could revaporize as a result of continued decay heating; fission product compounds would move to cooler surfaces.²³

Materials not retained in the reactor coolant system could escape to containment or other structures as aerosols borne with the escaping mixture of hydrogen and steam. There, the aerosols would be physically attenuated, principally by agglomeration and settling.

Additional aerosols containing fission products could be generated if an accident proceeds to the reactor vessel penetration stage. Aerosol-generating mechanisms include ejection of degraded core material from a pressurized reactor vessel, energetic interaction of hot core debris with a water pool below the vessel, and degradation of the concrete base mat by hot core debris.²³

3.3.5 Containment Effects

Containment is a key factor in determining if a severe accident results in a significant source term. If containment is not breached, the leakage of radioactivity will be inconsequential.

In some circumstances, engineered safety features can reduce the airborne concentration in containment in minutes, while natural aerosol depletion processes achieve similar reductions in several hours. If containment breaching is delayed more than a few hours after core degradation, the source term is much smaller as a result of these safety features and the effects of natural depletion processes. Significant release of radionuclides to the environment would occur only if the containment were breached in the first several hours after severe fuel damage.²³

Containments are required by regulation to be able to accommodate design-basis accident pressures and temperatures. Higher pressures and temperatures, however, can be imposed by severe accidents. These accidents could include steam explosions, steam pressure pulses, hydrogen burns, and long-term overpressure caused by steam production or the buildup of noncondensable gases such as hydrogen or carbon dioxide. Steam production by decay heat is an important heat-loading mechanism. Containment integrity is protected by engineered safety features such as containment sprays or atmosphere coolers. These systems are designed to extract the heat load; therefore, their failure must be postulated before a significant long-term pressure challenge to containment can be assumed. Passive heat transfer through the containment walls is insufficient to prevent long-term pressure buildup caused by decay heat.²³

Excessively high temperatures combined with elevated pressure can cause breaching of containment seals and penetrations, which are likely sources of containment openings in severe accidents. Another possible source is a procedural failure to close valves or other openings that penetrate containment.²³

Containments generally are partially or entirely surrounded by auxiliary structures, such as the service building or turbine hall of pressurized-water reactors or the reactor building of boiling-water reactors. Containment breaches are likely to lead into such structures, whose large volumes and surface areas provide opportunities for additional natural depletion processes to reduce leakage to the environment.²³

3.4 MODELING AND COMPUTER CODES

In the analysis of potential severe reactor accidents, modeling and computer calculations have been widely used. Computer programs (codes) have been substantially developed since WASH-1400 was published almost a decade ago, and many computer codes have been developed or improved recently.

By modeling (1) the fission product release during the fuel melting process, (2) the behavior and transport of fission products and aerosols in the reactor vessel, primary cooling system, and containment, and (3) the effect of engineered safety features, the computer programs can calculate the release from the containment to the environment.

The codes are usually arranged in suites, with each element in the suite addressing some part of the sequence. Output from one code generally serves as input for another code covering the next part of the sequence.

The codes are based on established physical and chemical laws and principles. However, approximations and applicability limitations often simplify the codes and reduce the time and expense of computer calculation. Experimental data can provide information of use to the code developer and can be used to check how successfully a given code or group of codes represent reality.

The performances of different codes used to predict the same part of a sequence can be compared by using each code to calculate the results for a standardized problem. Limited experience with verification by using such standardized problems and validation by predicting experimental results indicates that there is a basis for confidence in severe-accident analyses performed by knowledgeable analysts using current computer codes.²³

Different suites of computer codes have been used by different organizations involved in source term reevaluation, notably Battelle and IDCOR. Computer codes used by Battelle (under contract to NRC) are described in greater detail in Secs. 4.1.2.1-4; codes used by IDCOR are described in Sec. 4.1.2.7. Areas of disagreement in modeling and computer codes are discussed further in Sec. 5. Reference 42 lists various codes used in source-term calculations and gives brief code descriptions, contacts, and statuses.

3.5 SEVERE-ACCIDENT SEQUENCES

Much recent work has focused on the analysis of severe accident sequences. These sequences are chosen for analysis because they present the most-severe consequences and exemplify phenomena important in understanding the chemical and physical processes that determine fission product behavior in severe accidents. This work is an extension of the methodology advanced by WASH-1400 in 1975.

Most analysts have recently focused their attention on four families of accident sequences: large- and small-break losses of coolant, transient-initiated accidents (such as station blackouts), and containment bypasses. Many scenarios derived from these four

families of accident sequences have been examined. The accident sequences previously believed to have serious potential consequences have been studied in sufficient detail to estimate source terms.

Appendix A summarizes reactor parameters for the reference reactors that have been most thoroughly studied and lists accident sequences studied for each of the reference reactors. A key to accident sequence designation is given in App. B. Brief descriptions of accident sequences are given in App. C, and narrative descriptions of selected accident sequences are given in App. D.

3.6 COMPARISON OF ACCIDENT SEQUENCE SOURCE TERMS

Many estimates and calculations of source terms have now been made for various commercial light-water reactor designs and severe-accident sequences. Table 1 compares some of the source terms that have been proposed or used for regulatory or other purposes. To gain insight into developments in source-term reassessment, it is helpful, where possible, to compare earlier source terms with those calculated more recently for the same or similar accident sequences at the same or similar reactors.

For example, Table S.1 presents source terms for a severe-accident sequence (station blackout, the TMLB'- δ sequence) at a Surry pressurized-water reactor.^{22,43} The source terms shown were reported in WASH-1400 and a more recent Battelle study sponsored by the NRC.^{1,34,35} The source term for this accident sequence is appreciably lower in the more recent calculation, especially the components for radioiodines (lower releases could be significant in reducing potential early deaths and thyroid cancers) and cesium (lower releases could be significant in reducing potential long-term ground contamination and resource privation).⁴⁴

Tables 2-5 provide similar comparisons of source terms for other severe-accident sequences at a pressurized-water reactor (Surry) and a boiling-water reactor (Peach Bottom). In most (although not all) instances, the newer source terms are smaller. Tables 6 and 7 compare, for pressurized-water reactors and boiling-water reactors, respectively, source-term components for iodine, cesium, and tellurium for various accident sequences at several plants.

TABLE 2 Comparison of Source Terms for the Severe-Accident Sequence AB- γ (loss of coolant and loss of AC power) for a Surry Pressurized-Water Reactor

Organization and Report No.	Study Date	Percentage of Core Inventory Released						
		Xenon-Krypton	Iodine	Cesium-Rubidium	Tellurium-Antimony	Barium-Strontium	Ruthenium	Lanthanum
NRC, WASH-1400 ^a	1975	90	70	50	10	6	2	0.2
Battelle, BMI-2104, Vol. 1 ^b	1983	100	30	30	20	3	0.2	0.2
Battelle, BMI-2104, Vol. 5 ^c	1984	100	6	6	14	10	0.2	5

^aRef. 1.^bRef. 34.^cRef. 35.**TABLE 3 Comparison of Source Terms for the Severe-Accident Sequence TW- γ ' (transient event and buildup of residual core heat) for a Peach Bottom Boiling-Water Reactor**

Organization and Report No.	Study Date	Percentage of Core Inventory Released						
		Xenon-Krypton	Iodine	Cesium-Rubidium	Tellurium-Antimony	Barium-Strontium	Ruthenium	Lanthanum
NRC, WASH-1400 ^a	1975	100	90	50	20	6	3	0.4
Battelle, BMI-2104, Vol. 2 ^b	1984	100	30	30	30	-	-	-

^aRef. 1.^bRef. 45.**TABLE 4 Comparison of Source Terms for the Severe-Accident Sequence TC- γ (transient event and failure to scram) for a Peach Bottom Boiling-Water Reactor**

Organization and Report No.	Study Date	Percentage of Core Inventory Released						
		Xenon-Krypton	Iodine	Cesium-Rubidium	Tellurium-Antimony	Barium-Strontium	Ruthenium	Lanthanum
NRC, WASH-1400 ^a	1975	100	2	7	30	0.4	2	0.4
Battelle, BMI-2104, Vol. 2 ^b	1984	100	20	20	20	-	-	-

^aRef. 1.^bRef. 45.

TABLE 5 Comparison of Source Terms for the Severe-Accident Sequence AB-c (loss of coolant and loss of AC power) for a Surry Pressurized-Water Reactor

Organization and Report No.	Study Date	Percentage of Core Inventory Released						
		Xenon- Krypton	Iodine	Cesium- Rubidium	Tellurium- Antimony	Barium- Strontium	Ruthenium	Lanthanum
NRC, WASH-1400 ^a	1975	40	0.3	0.08	0.1	0.009	0.007	0.001
Battelle, BMI-2104, Vol. 5 ^b	1984	100	0.005	0.005	0.004	0.005	0.00003	0.004

^aRef. 1.

^bRef. 35.

TABLE 6 Source Terms for Severe Accidents at Pressurized-Water Reactors with Pre-Existing Openings in the Containment

Plant and Accident Sequence ^a	Organization	Area of Pre-Existing Opening (m ²)	Percentage of Core Inventory Released to Environment		
			Iodine	Cesium	Tellurium
Category PWR-2 (WASH-1400)	NRC	-	70	50	30
Surry					
ABB	Battelle ^b	0.033	8.7	8.5	7.0
ABB	Battelle ^b	0.033	5.0	4.9	4.7
ABB	Stone & Webster ^c	0.033	4.7	4.6	3.2
ABB	Stone & Webster ^c	0.0093	1.2	1.2	8.9
TMLB	Stone & Webster ^c	0.093	1.5	1.3	6.4
TMLB	Stone & Webster ^c	0.0093	0.21	0.21	2.3
V	Battelle	0.019	41	40	12
V	Stone & Webster	0.019	1.0	1.0	0.65
French 900-MW reactor					
ABB	Commissariat a l'Energie Atomique ^d	0.005	5.2	5.2	-
TMLB	Commissariat a l'Energie Atomique	0.005	1.7	1.5	-
German 1300-MW reactor	Kernforschungszentrum Karlsruhe	0.071	0.64	0.69	-
Zion					
TMLB	IDCOR	0.051	1.0	1.0	0.03
V	IDCOR	0.0093	0.0080	0.0080	0.0080
Sequoyah					
S ₂ HFB	IDCOR	0.051	2.0	2.0	0.46
V	IDCOR	0.0093	0.010	0.010	0.010

^aAccident sequences are defined in App. B.

^bRepeat of first reported calculation with four-node model of containment.

^cRetention in reactor coolant system is taken from Ref. 35.

^dRetention in reactor coolant system is taken from Ref. 34; retention outside containment is neglected.

Source: Adapted from Ref. 23.

TABLE 7 Source Terms for Severe Accidents at Boiling-Water Reactors with Containment Breach before Core Degradation

Plant and Accident Sequence	Organization	Containment Opening		Percentage of Core Inventory Released to Environment		
		Area (m ²)	Time (h)	Iodine	Cesium	Tellurium
Category BWR-2 (WASH-1400)	NRC	-	30	90	50	30
Peach Bottom						
TCy'	Battelle	0.65	0.97	34	29	32
TCy	Battelle	0.65	0.97	20	16	21
TC	IDCOR	0.019	13.0	3.0	3.0	7.0
TW	Battelle	0.65	29.2	28	27	18
TW	IDCOR	0.0093	34	13	13	13
Swedish BWR-3000						
TC	ELSAM	0.50	-	0.70	0.020	0.020
-	ELSAM	0.50	33.6	0.70	0.020	0.020
Shoreham						
TC	Science Applications	-	0.5	7.3	2.5	<0.050
TW	Science Applications	-	24	0.020	2.5	<0.30
Grand Gulf						
TC	Battelle	0.093	1.33	10	3.5	13
TC	IDCOR	0.14	1.0	0.080	0.080	0.080
TQW	IDCOR	0.0093	40	0.030	0.030	0.020
TPI	Battelle	-	22.2	5.1	3.4	3.2
Browns Ferry, TW	Oak Ridge	-	-	0.20	0.0058	-

Source: Adapted from Ref. 23.

4 CURRENT ACTIVITY RELATED TO SOURCE-TERM REEVALUATION

This section describes current experimental and analytical activities that focus on more-realistic determination of source terms for serious reactor accidents. Studies concerned with overall analytical reevaluation of source-term values are discussed in Sec. 4.1, while supporting programs directed at experimental measurements and verification are discussed in Sec. 4.2.

4.1 REEVALUATION STUDIES AND INTEGRATION OF RESULTS

4.1.1 Overview of Current Studies

Many government, industry, and independent scientific and technical organizations, both in the United States and abroad, are collecting and reviewing the expanding body of source-term information. The goal of these efforts is to substantially improve source-term values or methodologies (Table 8 lists some major institutions involved in the United States) and to adopt these more realistic assessments of possible fission product releases and the associated consequences for future regulation of nuclear reactors.

Current data indicate that source terms vary for different accident sequences and different plant configurations.¹⁴ Consequently, in addition to the diverse projects on such topics as fuel melting and the mechanisms of fission product release (see Sec. 4.2), various groups have been reevaluating risk assessment analyses of individual nuclear plants to quantify the reductions in severe-accident consequences implied by more-realistic source terms.

For example, the Electric Power Research Institute has been reevaluating the consequences of the important accident scenarios for a Surry reactor, one of two plants used as case studies in WASH-1400.^{13,23} Probabilistic risk assessments based on the new source-term data will be compared with those from the WASH-1400 for the same reactor. The U.S. Nuclear Regulatory Commission has sponsored work at the Battelle Columbus Laboratories on risk assessment of relevant accident sequences at six representative plants: a pressurized-water reactor like the ones at Surry, which was used as a reference plant in WASH-1400; the Peach Bottom (also a reference plant in WASH-1400) and Grand Gulf boiling-water reactors with Mark I and Mark III containments, respectively; the Sequoyah and Zion pressurized-water reactors with ice condenser and large dry containments, respectively; and finally a boiling-water reactor with Mark II containment.^{34,35,45-48} Source terms for these plants have been calculated for selected risk-dominant accident sequences (sequences that result in highest risk to the public) with the use of improved methodology and models of fission-product release mechanisms. Even with this range of plants, it is difficult to apply these results to the wide range of containment designs already in use and, in particular, to the crucial question of probabilistic early containment failure.¹⁴

TABLE 8 Major Institutions Studying Source-Term Reevaluation

Organization	Major Activities
U.S. Nuclear Regulatory Commission (NRC)	Program sponsor and regulatory authority
Battelle Columbus Laboratories	Revised source-term calculations for 6 reference plants for NRC
Oak Ridge National Laboratory	Computer code validation for NRC
Sandia National Laboratories	Evaluation of computer code uncertainties and sensitivities for NRC
American Physical Society	Peer review of reevaluation studies for NRC
Industry Degraded Core Rulemaking Program	Industry sponsor of analytical studies and calculations of source terms for reference plants
American Nuclear Society	Source-term reevaluation emphasizing general and phenomenological considerations
Electric Power Research Institute	Experimental safety research and input to IDCOR study
Stone and Webster Engineering Corp.	Several specific studies including one on aerosol behavior; input to American Nuclear Society study

A parallel effort has been undertaken by the Industry Degraded Core Rulemaking (IDCOR) working group, under the aegis of the Atomic Industrial Forum.⁴⁹ The IDCOR contractors have been studying a group of reference plants largely overlapping those in the Battelle study.^{14,49} The working group briefed its sponsors of its results in October 1983, and had its first technical interchange with the NRC during a three-day session November 29-December 1, 1983, that emphasized core heating phenomena, including steam and hydrogen explosions.⁵⁰ Later sessions in January and March 1984 covered fission product release and transport, IDCOR computer codes, and IDCOR analyses of power reference reactors (Zion, Sequoyah, Peach Bottom, and Grand Gulf). Preliminary information on the results, which have not yet been released to the public, indicates that IDCOR source terms are generally much lower than those calculated at Battelle. One main difference between the studies lies in the assumptions made about the possibility of early containment failure.¹⁴

A special committee of the American Nuclear Society (ANS) has taken a somewhat different approach to determine severe-accident source terms. This

committee has been making a semiquantitative assessment of factors affecting fission product retention in a nuclear plant, with particular attention to containment systems and accident sequences. The committee released a draft of the ANS source-term report for committee use only in November 1983; the results of the ANS study were to be made public at a special session entitled "Report of the ANS Special Committee on Source Terms" during the American-Nuclear-Society/European-Nuclear-Society International Conference held in Washington, D.C., November 11-16, 1984.

In addition, evaluations of the above studies have been in progress. These include a peer review study of the Battelle work.⁵¹ In addition, the American Physical Society set up a study group to examine radionuclide release from severe accidents at nuclear power plants. This group has been examining the Battelle, IDCOR, and NRC work and related work from national laboratories, including experimental work in progress.⁵² The study group's report will in turn be examined by an American Physical Society peer-review group.⁵³ The Union of Concerned Scientists is monitoring source-term reevaluation.⁵⁴

Internationally, the expanding body of source-term information is being collected and reviewed by the International Atomic Energy Agency and the European Organization for Economic Cooperation and Development's Nuclear Energy Agency Committee on the Safety of Nuclear Installations.^{14,16,17,55} However, the industry and regulators in other countries are very aware of the major impact that the deliberations of the U.S. NRC will have on the final outcome of source-term reevaluations and are unlikely to adopt interim or revised source terms before the United States does.¹⁴

Two challenging aspects of the source-term effort are integrating the results from the many experimental programs around the world and applying the results to the regulation of commercial light-water reactors. Ultimately, the research will lead to regulatory source terms that more realistically reflect the potential consequences and risks to the public of a severe nuclear plant accident. But the translation of laboratory and test reactor data into new regulatory standards will require the continuing cooperation of government, industry, and the research and development community.

In late 1982, the NRC established the Accident Source Term Program Office to coordinate the technical assessments of fission product release and integrate those assessments into the regulatory framework.^{51,56} Moreover, the NRC issued tentative schedules for incorporating research results into policy development and reactor licensing.⁵¹ The NRC staff recommendations for revised source terms, however, might not be issued until the various studies now in progress are completed. However, the NRC staff will propose a source-term rule specifying the fission product release expected for severe accidents at all light-water power reactors now operating or under construction.⁵⁷ The NRC staff has apparently decided to leave resolution of regulatory issues until after source-term questions are resolved: the severe-accident debate thus will not be ended before 1985.⁵⁰

4.1.2 Individual Programs

4.1.2.1 NRC Source-Term Program

As a result of the accident at Three Mile Island and subsequent reevaluation of nuclear regulatory processes, the NRC initiated, on October 2, 1980, a "long-term rulemaking to consider to what extent, if any, nuclear power plants should be designed to deal effectively with degraded core and core melt accidents."⁴⁹ As a consequence, reassessment of source terms for severe accidents has been the subject of intense activity by the NRC. The NRC is addressing the issue only for commercial light-water power reactors. Quantity, characteristics, and timing of radionuclides released to the environs after an accident are being assessed. The NRC established the Accident Source Term Program Office to focus and direct the commission's efforts in this area and attempt to resolve the issues.^{52,58} The NRC has also sponsored several series of reactor safety meetings and peer group reviews (see Secs. 4.1.2.2 and 4.1.2.5).⁵⁹

Major input into NRC source-term reassessment comes from studies conducted at Battelle Columbus Laboratories, the results of which are published in Refs. 34, 35, 45-48, 60; Battelle work is discussed in more detail in Sec. 4.1.2.2.

To calculate radionuclide and aerosol releases, Battelle researchers developed computer codes and applied them to specific plants and selected accident scenarios. The codes mathematically model fission product release during the fuel melting process; the behavior and transport of fission products and aerosols in the reactor vessel, primary cooling system, and containment; the effect of engineered safety features; and the release from the containment to the environment.

The computer codes used (discussed in Sec. 4.1.2.3) are based on mechanistic, phenomenological models of processes when such models are available and on simplified approximations of other important natural physical and chemical processes. Radio-logically important elements such as iodine, cesium, and tellurium, in vapor or particulate states, are emphasized.

Battelle has analyzed and reported on five reference plants (Surry, Zion, Peach Bottom, Grand Gulf, and Sequoyah) and is analyzing one additional Mark II boiling-water reactor reference plant.^{34,35,45,47,48,60} A number of key technical issues have been identified during the peer review of the Battelle work. Some of these issues have been addressed by the NRC source-term contractors (including Oak Ridge and Sandia National Laboratories) within the limitations of current technology. Special task groups have been formed from the NRC staff to resolve certain other issues with contractor assistance, while in some cases the ongoing NRC research program has been redirected or augmented to provide additional analysis or methodology improvement. Some issues will not be resolved until the data base from current research is developed sufficiently to confirm methodologies used in source-term reassessment findings.⁵⁹

Validation of the source-term data base and computer codes has been conducted for the NRC by Oak Ridge National Laboratory (Sec. 4.1.2.3). The sensitivity and uncertainty analyses of results from Battelle have been the responsibility of Sandia National Laboratories (Sec. 4.1.2.4).

Integration of all the NRC source-term reassessment projects will culminate in a report prepared by the NRC staff, NUREG-0956. After the various supporting tasks are completed (analyses of containment performance, uncertainty and sensitivity, and thermal hydraulics), the staff will appraise the risk and regulatory significance of reassessed source terms. This appraisal will be based on NRC contractor analyses and findings, individual technical peer reviews of various source-term reassessment studies, and review of the IDCOR work.¹⁴

On the basis of the staff appraisal at the end of 1983, the broad-based scientific peer review by an independent scientific organization, and the review of the American Nuclear Society study, Battelle was directed to proceed with refined source-term analyses for specific plants and accident sequences in January 1984. The NRC has continued preparing its appraisal, factoring in results from the new Battelle analyses. Its final appraisal will be published in NUREG-0956 after public comments and the findings of the independent scientific peer review are received.^{51,52} Although NRC originally intended to release this report by December 1984, release is now expected to be delayed, probably until May 1985.

4.1.2.2 Battelle Source-Term Program

The Battelle Columbus Laboratories source-term program has provided the major initial input to the NRC's current source-term reassessment. The Battelle work has been documented in Refs. 34, 35, 45, 47, 48, and 60, which are all volumes of Battelle report BMI-2104. Some volumes of this set (Vols. 1-6) have been released in bound draft form and some (Vol. 7 and possibly further revisions of previous volumes and supplements) have not yet been released.

The purposes of the Battelle report are to:⁶¹

1. Develop updated fission-product source terms for five particular types of nuclear power plants and for several accident sequences that present a range of conditions. The source terms are to be based on analyses of fission product release from fuel, transport, and deposition. These analyses are to be performed with a consistent step-by-step approach and improved computational tools.
2. Determine the effects on calculated fission-product release estimates of major differences in input parameters for plant design and accident sequences.
3. Provide in-plant time- and location-dependent distribution of fission product mass.

In the overall Battelle approach, the specific plants and accident sequences were selected for consideration, then analyzed for fission product release from fuel, transport, and retention to predict fission product release to the environment for these specific

cases. The approach comprises a series of steps in the overall analysis and the results are specific to an individual set of accident conditions for each reactor. Each successive step is based on results from analysis of the previous step.⁴⁶

The specific plants and sequences chosen for study are as follows:

- Surry -- pressurized-water reactor, large dry containment; accident sequences AB, TMLB', S₂D, and V.
- Peach Bottom -- boiling-water reactor, Mark I containment; accident sequences TC, AE, and TW.
- Grand Gulf -- boiling-water reactor, Mark III containment; accident sequences TPI, TQUV, TC, and S₂E.
- Sequoyah -- pressurized-water reactor, ice condenser containment; accident sequences TML, TMLB', and S₂HF.
- Zion -- pressurized-water reactor, large dry containment; accident sequences TMLB' and S₂D.

Appendix B gives a key to accident sequence designations. The accident sequences were selected to include those with high risk, severe consequences, and a wide range in physical conditions.

After sequences were selected, plant design data were collected and thermal hydraulic conditions were analyzed for each sequence. Overall thermal hydraulic conditions were estimated on a time-dependent basis with the MARCH code, and detailed thermal hydraulic conditions for the primary system were estimated with the MERGE code, which was developed specifically for use in this program. The CORSOR code was used to predict time- and temperature-dependent mass releases of radionuclides from the fuel within the pressure vessel. Radionuclide releases during core-concrete interactions with the melt were provided by Sandia National Laboratories with its code VANESA.⁴⁶ Figure 1 is a flowchart of the Battelle codes for release, transport, and retention calculations.

With the thermal hydraulic conditions predicted by MARCH and MERGE and the radionuclide release rates predicted by CORSOR as input, the TRAP-MELT 2 code was used to predict vapor and particulate transport in the primary coolant circuit. Transport and deposition of radionuclides in the containment were calculated with the NAUA-4 code. Aerosol scrubbing in pressure suppression pools was predicted with the SPARC code developed by Battelle/Pacific Northwest Laboratory.⁴⁶

The calculations performed were "best estimates" and used input derived from experimental measurements to the extent possible. The Battelle group recognized that the use of mechanistic models to compute radionuclide release and transport is subject to many uncertainties of various magnitudes and importance. The study did not quantify uncertainties in individual parameters and hence the overall importance of such uncertainties has not been assessed. However, the sensitivity of results to a few specific parameter variations was evaluated.⁴⁶

FIGURE 1 Flow Chart of Battelle Computer Codes for Release, Transport, and Retention Calculations

The calculations show that fission product release to the environment depends strongly on the specific plant design and accident sequence. Also, assumptions of time estimated for containment failure were of major importance. The calculated release fractions ranged from values near those given in WASH-1400 for the appropriate category to much lower values.⁴⁶

As an example of release fractions, the largest source term in the earlier Battelle study of a Surry pressurized-water reactor results from an accident sequence referred to as TMLB' (transient event and loss of primary-system heat removal).³⁴ The source term derived by Battelle differs from that in the WASH-1400 in several respects. For example, 100% of the core inventory of noble gases (xenon and krypton) was considered to be released to the environment in the Battelle study versus 90% in the WASH-1400. Also, WASH-1400 assumed iodine was in the elemental (I_2) gaseous form; Battelle estimated that only 0.05% of the iodine was gaseous and treated the remainder of the iodine as a cesium iodide aerosol. Both studies initially reported that 70% of the iodine was released to the environment. However, Battelle reported a much lower value (5%) in its recalculations for the Surry plant published in Ref. 35. Battelle also originally calculated that releases of the cesium-rubidium and tellurium-antimony groups of fission products to the environment were somewhat greater than those calculated in WASH-1400, but later recalculated lower values.^{34,35} The percentages of the core inventory of the remaining fission product groups released to the environment as recalculated were smaller than or comparable to WASH-1400.

The following BMI-2104 bound draft reports are available: *Volume I — PWR, Large, Dry Containment (Surry)*; *Volume II — BWR, Mark 1 Design (Peach Bottom)*; *Volume III — BWR, Mark III Design (Grand Gulf)*; *Volume IV — PWR, Ice Condenser Containment Design (Sequoyah)*; *Volume V — PWR, Large, Dry Containment Design (Surry Plant Recalculations)*; and *Volume VI — PWR, Large, Dry Containment Design (Zion Plant)*. Peer review comments on the Battelle studies and a supplementary volume with limited calculations on a boiling-water reactor reference plant with Mark II containment are expected to be available during late 1984 or in 1985.

4.1.2.3 Source-Term Program at Oak Ridge National Laboratory

Oak Ridge National Laboratory has contributed to the NRC reassessment activity in source-term code validation, as well as other areas.⁶²⁻⁶⁴ (Other Oak Ridge experimental source-term projects are discussed in Sec. 4.2.) Because source-term determination relies so heavily on large, complex computer codes, the credibility of source terms rests on the assessed validity of these codes. Therefore, one task performed by Oak Ridge for the NRC's Accident Source Term Program Office has been to review the validation of the source-term codes used in the Battelle study. Code validation at Oak Ridge did not include a rigorous sensitivity and uncertainty analysis; that work has been conducted at Sandia National Laboratories. Oak Ridge plans called for providing the following information for each of the codes used (ORIGEN-2, MARCH 2.0, CORSOR, MERGE, TRAP-MELT, CORCON, VANESA, and NAUA-Mod4):^{65,66}

- Description of the physical models in the code,

- Discussion of the status of documentation,
- Compilation of the required inputs and major outputs,
- Discussion of the implied and explicit assumptions and any perceived limitations in the code,
- Identification and assessment of the data bases for the inputs, prefixed parameters, and internal phenomenological models,
- Description of the status of verification and quality assurance,
- Assessment of validation needs and description of NRC programs that are addressing those needs, and
- Discussion the validity of the code as used in the study.

A real problem in assessing the validation of any code is the lack of (1) a formal definition of a "validated" code and (2) any agreed-upon and documented formal validation procedures.⁶⁵

Overall, Oak Ridge judged that the codes lack experimental validation and treat some phenomena in a crude simplistic fashion. Nevertheless, they are basically sound and use good fundamental approaches. Some important phenomena, however, are left untreated. Some specific areas identified as inadequately treated by the codes included:⁶⁵

- Details of fuel melting and slumping behavior;
- Fuel debris development, characteristics, and influence on thermal hydraulic behavior;
- Chemical and physical interactions of various fission product species with surfaces;
- Thermal hydraulic response of the primary system, particularly in the upper plenum regions;
- Inclusion of simultaneous coupling of aerosol transport behavior and thermal hydraulic behavior;
- Assessment of the chemical forms of all the fission products;
- Inclusion of fragmentation of both fuel and water on energetic expulsion from the primary system.

Reviews by Oak Ridge of the status of code validation have been given to the Battelle peer review group⁶⁶ and the American Physical Society source-term study group.⁶⁸ The

results of the validation status review will be published as ORNL/TM-8842; a preliminary limited-distribution draft has appeared.⁶⁷ A final report is expected in November 1984.

4.1.2.4 Sandia Source-Term Program

Evaluation of the uncertainties and sensitivities associated with the source-term values developed in the NRC's interim source-term study (as presented in the Battelle study report) has been the responsibility of Sandia National Laboratories.⁶⁹ The quantitative uncertainty estimation for the source-term (QUEST) study was designed to determine the source-term uncertainty in two categories. The first category concerns alternate, but reasonable, inputs to the suite of computer codes used. These codes include MARCH, MERGE, CORSOR, TRAP-MELT, CORCON, VANESA, NAUA, and SPARC (see Fig. 1). The second category of uncertainty concerns phenomena that are known but either omitted or poorly modeled in the suite of codes used.

A similar approach was taken to determine the source-term uncertainty for both categories. First, sources of uncertainty and key input parameters were identified. Second, the ranges over which each source of uncertainty or input parameter might vary were determined. Third, the sources of uncertainty were combined and propagated to yield the net uncertainty in the radionuclide release.

Some sources of uncertainty were identified from those noted by the peer review of the Battelle study, as well as those noted in other uncertainty studies. In addition, sensitivity studies were reviewed (or performed) to help determine which sources of uncertainties could strongly influence the amount of released radionuclides.

Once key sources of uncertainty were identified, the range over which they might vary were determined. These ranges were determined by the spread in available data and in predictions of defensible models. For example, the uncertainty range in the coefficients for the release of cesium at a given temperature can be determined from the large amount of data available on cesium release. However, the uncertainty in the time and temperature history of the fuel (which also affects cesium release) can only be determined by calculations based on various assumptions. The range for each key source of uncertainty was then documented and subjected to peer review. However, because time for this study was limited, the confidence level for each range was not stated, nor were subjective probability distributions constructed.⁶⁹

Combining and propagating the input uncertainty ranges is not an easy task; the technique used must allow for synergisms and avoid inconsistencies. The Monte Carlo method (in which values for each input parameter or process are chosen at random from within their ranges) can meet these criteria if each set of values chosen is internally consistent. However, the Monte Carlo method requires many full calculations and was therefore not acceptable for such a short study. The method to be used might be called a "selective" Monte Carlo method. The set of parameter or process values for the calculations is chosen to yield a large range in the calculated radiological source term. Sensitivity studies and scoping calculations can guide the choice of input values. Only a very few full calculations were to be performed. The resulting range in the calculated source term would not be the greatest possible but should be representative of the source-term uncertainty and would not require many full calculations.

The Sandia study was to use the same source-term groupings the Battelle study used (iodine, cesium, tellurium, refractory fission products, and nonradioactive aerosol materials). The range of airborne materials within containment as a function of time was to be displayed with containment failure time as an independent parameter. In addition, the range in the total amount of radiation released from containment was to be determined for two specific failure times (early and late) and was to include effects after containment failure such as resuspension or agglomeration and settling of aerosols. Because the study time was limited, only a few plants and sequences were to be considered -- in particular, Surry with the accident sequences TWLB' and S₇D, and Peach Bottom with the accident sequence TW. These accident sequences are considered in the Battelle study and dominate the risk for these plants.⁶⁹

This quantitative uncertainty estimation for the source-term study was constrained by the original six-month time limit, which was subsequently extended. Some of the initial uncertainty analyses were presented and discussed at the third peer review meeting on the Battelle study (held July 28-29, 1983, in NRC headquarters), the Eleventh Water Reactor Safety Research Information Meeting (held in Gaithersburg, Maryland, October 24-28, 1983), and the American Nuclear Society Topical Meeting on Fission Product Behavior and Source Term Research (held in Snowbird, Utah, July 15-19, 1984).^{69,70,71} Detailed studies of uncertainties for the Surry TWLB' accident sequence including quantitative results were reported at the latter two meetings.^{70,71} While quantitative uncertainty values differ for different radionuclides and accident sequences, uncertainties generally have been reported to be mostly downward.⁷⁰ By the end of September 1984, draft versions of Volume I of the QUEST study (an executive summary) and Volume II (for the S₇D accident sequence) had been delivered to NRC for internal use, while reports on two additional sequences had not yet been written.

4.1.2.3 Source-Term Program of the American Physical Society

In response to NRC's need for a broad-based peer review by an independent scientific organization, the American Physical Society appointed a study group to review radionuclide release from severe accidents at nuclear power plants.⁵² This study group, which is chaired by Richard Wilson of Harvard University, consists of 12 members who have diverse technical backgrounds and come from different institutions.⁵³ The American Physical Society has also appointed a peer review committee charged only with evaluating the results of this study group.⁵³ The NRC has provided a grant to the American Physical Society for this work.⁵⁴

The American Physical Society group focused on the fundamental physics and chemistry of the radionuclide release, making an effort to define the governing physical phenomena and the important controlling parameters.⁵⁵ In particular, the study was proposed to include (1) critical review of NRC-supported experimental and theoretical work that has already been completed in this area, including analyses that have been performed for specific reactors; (2) evaluation of the extent to which more-accurate estimates of radionuclide releases are now possible; (3) identification of areas in which the scientific basis for such estimation is now inadequate; and (4) indication of the degree to which presently planned or additional research efforts can be expected to

rectify these inadequacies. The study group also examined other work relevant to source-term reassessment.⁵²

The American Physical Society study group was originally given 18 months to complete its work. The group started its work in September 1983 and has had a series of meetings since then.^{52,72,73} A meeting during late July 1984 in Snowmass, Colorado, was held to draft the group's report, but completion of the report has been delayed. A draft report is scheduled for release in 1985, possibly January, and a final report is anticipated to be released in April 1985.⁷²

4.1.2.6 Stone and Webster Source-Term Program

Stone and Webster Engineering Corporation (SWEC) has conducted a fairly extensive source-term program since 1981. The SWEC program is characterized by a strong systems engineering orientation and advanced analytical techniques.⁷⁴

A number of SWEC investigations have assessed the radiological consequences of postulated nuclear power plant accidents.^{28,29} These assessments have generally been based on studies of accident consequences rather than probabilities and have dealt with the effects on radiological exposures of reduced source terms. On the basis of these assessments, SWEC has proposed greatly reduced interim source-term values (see Table 1). It has concluded that previous source-term assumptions, while reasonably accurate for the noble gases, lead to significant overestimation of release concentrations of other fission product isotopes.

The SWEC investigations supporting these reduced source terms have emphasized thermal hydraulic analysis of the containment, including size of containment opening and timing of containment opening for a single-node containment model; multicompartment containment models; thermal hydraulic analysis of structures outside the containment in containment bypass scenarios; aerosol calculations; and investigation of additional phenomena, including diffusiophoresis and suspended liquid effects.⁷⁵

Generally, work done by SWEC includes analyses of thermal hydraulics and containment structural integrity, parametric studies of fission product retention, selection of representative accident sequences, comparison of aerosol codes, studies of revaporization; studies of containment bypass sequences, development of a vapor-and-aerosol mass-transport model that considers chemical and decay heat effects, development of a coupled analysis of thermal hydraulics and fission product transport in reactor coolant systems, and consequences studies to identify important nuclides.⁷⁴

Personnel at SWEC have participated in the NRC Accident Source Term Program Office peer reviews, the IDCOR Expert Review and Special Review Groups, the peer review of the EPRI reevaluation of WASH-1400 analyses (see Sec. 4.1.2.9), the NRC's Containment Loads Working Group, the IDCOR Policy Group, the Atomic Industrial Forum Emergency Preparedness Subcommittee and its NUREG-0396 Working Group, and the Industry Ad Hoc Committee on Source Terms Policy, and have contributed to the report of the American Nuclear Society Special Committee on Source Terms. In addition, SWEC researchers have presented their results to the American Physical

Society Study Group on Radionuclide Release from Severe Accidents at Nuclear Power Plants and have published many reports in this area.^{37,76-85}

4.1.2.7 IDCOR Source-Term Program

In response to the NRC-initiated process aimed at reviewing existing regulations to determine what, if any, changes should be made to reflect the possibility of core damage occurring in commercial nuclear power plants, the nuclear industry developed the Industry Degraded Core (IDCOR) program to prepare an industry position on the many technical issues surrounding the degraded core question and to reach satisfactory resolution with the NRC on this matter.⁸⁶ The IDCOR Program is a multimillion-dollar program sponsored by the nuclear industry. Participants include 48 utilities, 11 design or architect-engineer firms, and representatives from Sweden, Japan, and Finland. The program is directed by a policy group comprising representatives of the sponsoring organizations and operating under the corporate auspices of the Atomic Industrial Forum. The program's purpose is to develop a comprehensive, integrated, technically sound position on whether changes in regulations are needed to reflect degraded core and core melt accidents. The management structure is given in Ref. 49.

Although in the past, several commercial applicants for nuclear power plant licenses have had to consider specific technical questions about degraded cores, the industry anticipated that NRC's degraded core rulemaking might require all licensees of light-water reactors to consider degraded core accidents as part of the design basis for their plants. Further, NRC licensing actions -- the rulemaking on degraded core issues as well as actions at specific plants -- suggested that NRC might be moving in the direction of requiring significant additional devices to mitigate the effects of degraded core accidents. By requiring such devices, NRC would, to some extent, change the current design of nuclear plants.⁴⁹

The IDCOR program plan was developed to identify the integrated efforts necessary to generate sufficient technical information on key degraded-core issues. The overall objective of the IDCOR program has been to provide a strong, documented technical base and develop integrated technical and legal options that support industry participation in the NRC degraded-core rulemaking process.⁸⁶

The following strategy was structured to achieve the program objective -- that is, to assure that a rule, if developed, would be based on technical merits and would be acceptable to the nuclear industry:^{49,86}

- Conduct the technical program,
- Develop integrated technical and legal options,
- Work with NRC to reach agreement on the technical aspects of degraded core issues and any proposed rulemaking process so that future rulemaking will be based on technical merit and not develop into a protracted procedural battle.

The policy guiding the IDCOR program's development of a well-documented and technically sound position on degraded core issues is guided by the following four principles to ensure a realistic appraisal:⁴⁹

- Review and use existing information to the maximum extent possible;
- Develop programs enhancing technical understanding and communications;
- Use realistic, rather than conservative, engineering approaches in technical analyses; and
- Use an independent expert review process.

The use of realistic, rather than conservative, engineering approaches in technical analyses balances the assessment of accident concerns and fosters a focus on accident prevention as well as a better understanding of the potential for and benefit of accident mitigation considerations. Use of conservative and bounding analyses might narrow attention to the very low probability events and unreasonably overestimate their significance.⁴⁹

The IDCOR program consists of a logical sequence of analytic activities as defined by the 24 technical tasks listed in Table 9. Specifically, the IDCOR program approach has been to:

- Develop generic evaluative criteria based on safety goals,
- Select a set of representative reference plant designs,
- Identify the dominant accident sequences that can result in degraded cores if they are allowed to proceed unchecked,
- Realistically characterize reactor behavior involved in these dominant sequences, and
- Relate results to the evaluative criteria.

The IDCOR program was intended to undertake generic analyses to the greatest possible extent. However, many analyses, particularly those pertinent to an assessment of preventive or mitigative actions, must take into account specific features of design, equipment, and operation. For this reason, the IDCOR program selected specific plant designs for analyses that represent the categories of existing plants. The reference plant designs were selected from the following:

- Pressurized-water reactor with large containment,
- Pressurized-water reactor with ice condenser,

TABLE 9 IDCOR Technical Tasks and Objectives

Task	Objective
1. Evaluate safety goals	Use a quantitative safety goal to address degraded core rulemaking
2. Establish ground rules for evaluation of degraded conditions	Establish ground rules for the analyses of selected IDCOR accident sequences
3. Select dominant accident sequences	Determine the dominant contributors to plant risk so possible risk reductions can be examined
4. Select phenomenological sequences affecting containment	Provide a mechanism for examining phenomenological effects on dominant accident sequences
5. Determine effects of human error on dominant accident sequences	Examine the impact of human errors on dominant accident sequences
6. Establish risk significance profile for engineered safety features and other equipment	Establish a risk profile for engineered safety features and other equipment
7. Establish baseline risk profile for current-generation plants	Establish a current risk profile for light-water reactors
8. Determine effect of post-Three-Mile-Island changes on the overall risk profile	Examine the risk reduction resulting from post-Three-Mile Island changes
9. Identify preventive methods to arrest sequences of events before core damage	Identify potential improvements in design and operation that would reduce the likelihood of core damage
10. Examine containment structural capability	Establish a uniform, realistic basis for determining containment structural capability
11. Predict fission product liberation, transport, and inherent retention	Predict fission product release from fuel, transport and distribution within containment, chemical and physical form, and ultimate disposition during dominant accident sequences

TABLE 9 (Cont'd)

Task	Objective
12. Predict hydrogen generation and burn	Predict hydrogen generation and distribution and the subsequent hazard from over-pressure and combustion
13. Evaluate hydrogen burn control	Evaluate means to prevent, suppress, or control hydrogen burn
14. Predict steam overpressure phenomena (in-vessel containment)	Predict the rate of steam generation during dominant accident sequences and assess steam explosions
15. Predict core debris behavior and cooling	Predict the behavior of core materials and their interactions with structural materials during dominant accident sequences and determine coolability limits
16. Integrate model and analysis	Integrate the results of other IDCOR sub-tasks into a comprehensive capability for predicting plant behavior during dominant degraded-core accident sequences
17. Evaluate equipment survivability in a degraded core environment	Identify a minimum set of equipment and evaluate its capability to survive a degraded core environment
18. Evaluate atmospheric and liquid pathway dose	Identify the most appropriate models for evaluating radiological consequences of dominant sequences and perform analyses
19. Evaluate alternative containment systems	Review ongoing studies of alternative containment systems and evaluate the potential for risk reduction and undesirable side effects
20. Evaluate core retention devices	Review ongoing studies of core retention devices and evaluate the potential for risk reduction and undesirable side effects
21. Examine risk reduction potential	Examine reductions in plant risk due to current activities and possible future actions
22. Identify safe stable states after core degradation	Identify any potential safe stable states that may provide a means of terminating an accident

TABLE 9 (Cont'd)

	Task	Objective
23.	Perform integrated analysis of containment systems	Perform realistic analyses of thermal, fluid, pressure, and fission-product loads on containment systems
24.	Characterize operational aspects of accident management and control	Characterize the potential for operator response in managing degraded core conditions

Source: Adapted from Refs. 49 and 86.

- Pressurized-water reactor with special characteristics (such as Babcock-and-Wilcox once-through steam generators or Combustion Engineering lower head design of reactor pressure vessel)
- Boiling-water reactor with Mark I containment,
- Boiling-water reactor with Mark II containment, and
- Boiling-water reactor with Mark III containment.

Reference plants studied by IDCOR were mainly the same as those studied by Battelle (Sec. 4.1.2.2). In most cases, however, different accident sequences were studied by IDCOR and Battelle. The reference plants and accident sequences are detailed in App. A.

The 24 technical tasks listed in Table 9 constitute the foundation of the IDCOR technical program. An essential part of the program is the integrated modeling. Available codes were assessed, their uses were defined, and related activities were tracked so the results of other IDCOR subtasks could be integrated into a comprehensive accident analysis capability for predicting plant behavior during dominant degraded-core accident sequences.

Support was be provided for the development of the IDCOR Executive Analysis computer program for integrated analysis of transients associated with dominant sequences. Existing codes and ongoing developments (in particular activities supported by EPRI's Nuclear Safety Analysis Center, such as comparisons of the MARCH and KESS codes, upgrading of MARCH, and development of the TMI code) that could be used within the overall IDCOR integrated analysis were identified. Information was exchanged between IDCOR and Nuclear Safety Analysis Center on analyses, and NRC and foreign programs were monitored.

The IDCOR Executive Analysis computer program was developed for estimating plant behavior under severe accident conditions. It includes provisions for modeling both pressurized- and boiling-water reactors. The executive code was intended to cover three phases including (1) intact geometry core heat-up, (2) in-vessel material migration and (3) containment analyses. The analysis package was to use the best available phenomenological models and data on parallel activities (e.g., the MARCH upgrade and TMI code development). Additional phenomenological models developed in IDCOR tasks (tasks 11, 12, 14, and 15 of Table 9) were also to be integrated into the Executive Analysis program.⁴⁹ The simplified results of the evaluation of containment structural capability (task 10 of Table 9) were also to be incorporated in the model.⁴⁹

The IDCOR code suite is similar to that described earlier for the Battelle code suite. However, the IDCOR suite has fewer individual code members (see Fig. 2); it comprises only the Modular Accident Analysis Program (MAAP), Fission Product Release and Transport (FPRAT) program, and fission product transport and retention program (RETAIN).⁸⁷

The MAAP code was developed for analyzing severe accidents at both pressurized- and boiling-water reactors. Both versions are thermal hydraulics codes that describe responses of light-water reactor systems to accident-initiating events up to either a safe, stable, coolable state or containment breaching and depressurization. The MAAP code roughly corresponds to MARCH in the Battelle suite, together with appropriate parts of MERGE, CORCON, and CORSOR (see Sec. 4.1.2.2).

FIGURE 2 Flow Chart of IDCOR Computer Codes for Release, Transport, and Retention Calculations

The FPRAT code calculates the release of fission products from the fuel matrix, release of vaporized core materials, and transport of these vapors to the upper regions of the reactor vessel. The thermal hydraulics parameters from MAAP output -- namely pressure, gas flow rates, ratios of hydrogen to steam in the gas, and temperatures of gas and other materials, all as functions of time -- are used to calculate release rate of fission products from the fuel. The FPRAT code is the analog of CORSOR in the BMI suite for core material behavior before fuel melting and slumping.

The RETAIN code tracks the transport of fission products from their initial release from the fuel, through the primary system, and to their ultimate retention inside reactor building structures or escape to the environment. The output from FPRAT is used as input to RETAIN along with the temperature histories calculated by the thermal hydraulics model in MAAP. The RETAIN code corresponds approximately to the TRAP-MELT/NAUA part of the Battelle code suite.

The CRAC-2 code is used to determine the environmental consequences of severe accidents. The characteristics of the source terms from the output of RETAIN are used as input to CRAC-2, which is a revision of the CRAC code that was developed for WASH-1400.

A key component of the IDCOR effort has been the phenomenology work of Fauske Associates Incorporated. This work is based on use of an integrated code that provides for accurate transmission of fission product composition and amounts from one phase of containment to another. It therefore deals not only with fission products released from the fuel but also with their transition from the reactor vessel to the primary system and from there to containment. Input for each phenomenon is based on EPRI (and other) source-term research.

A review of the completed Fauske analysis determined that the latest experimental data were taken into account and the modeling is defensible. The work passed a peer review at EPRI on November 1-3, 1983. The Fauske results indicate that overall retention factors are quite large. For accident sequences in which the containment is not bypassed, the Fauske retention factor for iodine was very large and the retention factors for cesium and tellurium were 30 or greater. Even when the containment is bypassed, these retention factors were found to be 10 or more and could result in no early fatalities. The Fauske study may also show that events such as metal-water reactions and hydrogen generation are not likely to threaten the building integrity.

The Fauske results for IDCOR reportedly indicate that the source term for any credible set of failure assumptions is so small that NRC siting requirements need to be changed. The Fauske staff reportedly believe that the IDCOR work is unassailable and that NRC needs to modify its source term accordingly.⁸⁸

Many reports have been prepared under the auspices of IDCOR. These are listed for the convenience of readers in the bibliography. The final IDCOR report has not yet been published, but should be available in December 1984.

4.1.2.8 Source-Term Program of the American Nuclear Society

The American Nuclear Society has been involved in source-term assessment since 1982 and is scheduled to complete its work by the end of 1984. The huge disparity between predicted and actual radioactive releases from the accident at Three Mile Island 2 prompted substantial government and industry efforts to devise more-realistic source-term estimates. Most of these efforts, though incomplete, indicate that source terms should be reduced, although there has been some disagreement on the size of the reductions.

Consequently, the American Nuclear Society chartered a Special Committee on Source Terms, chaired by W.R. Stratton, to achieve a consensus of the technical community on the methods and assumptions appropriate for use in determining source terms. The 20-member committee included representatives from the United Kingdom, France, Japan, the Commission of European Communities, and the United States.

This committee's task was to examine, review, and evaluate current knowledge about establishing source terms for particular accident sequences, including data on fission product release from fuel, transport to and retention within containment (or other buildings), and chemical form and behavior. The committee was to quantify the fission product retention associated with each of the various relevant phenomena, plant systems, and structures, making comparisons with the assumptions and conclusions of WASH-1400 and other related work as appropriate.

The scope of the committee's activities and its results are to be published in a report covering the following topics:

- Scope of study and history of source-term assumptions.
- Accident sequences for pressurized- and boiling-water reactors.
- Important radionuclides (factors determining importance; WASH-1400 classification of noble gases, halogens, alkali metals, tellurium, alkaline earths, noble metals, rare earths, and elements with refractory oxides; relative importance of radionuclides).
- Chemical and physical forms and interactions of fission products, aerosols, escape of materials from a degraded core, and transport and retention characteristics of reactor coolant systems and containments.
- Computer programs for accident analysis.
- Discussion and comparison of accident sequence calculations.
- Results of parametric studies by Stone and Webster Corp., IDCOR, NRC (Battelle), WASH-1400, and others.

The American Nuclear Society has been active in sponsoring numerous meetings on thermal reactor safety and source-term research, including an American Nuclear Society topical meeting on fission product behavior and source-term research in Snowbird, Utah, on July 15-19, 1984.^{10,11,89} A preliminary review of the results of the work of the Society's Special Committee on Source Terms was presented at the Fifth International Conference on Thermal Nuclear Reactor Safety sponsored by the European Nuclear Society and the American Nuclear Society in Karlsruhe, Federal Republic of Germany, September 10-13, 1984.²³ It is anticipated that a full report of the Special Committee on Source Terms results will be made at the American Nuclear Society/European Nuclear Society International Conference in Washington, D.C., November 11-16, 1984.

The committee's results to date were reviewed by W.R. Stratton.²³ The committee concluded that the state of knowledge and the analytical methods and assumptions on which current calculations of source terms are based have progressed far beyond those on which WASH-1400 was based. In general, an ample foundation has been provided to warrant reductions of the source-term estimates in WASH-1400 by more than an order of magnitude to as much as several orders of magnitude. The noble gases are exceptions because they are chemically inert and do not undergo the wide range of chemical and physical interactions that reduce releases of most fission products; however, their inert character also leads to low radiological consequences.

The committee also found that iodine will be released and transported predominantly as cesium iodide and cesium as cesium hydroxide. These species will form aerosols and be subject to aerosol depletion processes. In addition, they are highly soluble in water and can be irreversibly adsorbed onto metal surfaces. These characteristics result in greatly reduced releases compared to those in WASH-1400. This finding holds for all light-water reactors and all accident sequences.

The more-severe accident sequences developed in WASH-1400 and more-recent probabilistic risk assessment studies provide a sufficiently complete basis for in-depth analyses of source terms. These sequences include those most likely to result in large releases, and they involve the phenomena and processes that are considered to affect the escape and transport of fission products. Sequences and plant details are important in estimating plant-specific source terms.

If the containment is not breached, essentially no fission products are released; if containment breach is delayed more than a few hours after core degradation, the source term is greatly reduced, independent of the final size of the containment breach. The containment is less susceptible to early breaching than previously believed. Substantial evidence indicates that a high degree of confidence can be placed in calculations of light-water-reactor source terms.

The committee made several specific findings relative to containment. Three of the four causes of early containment breach are no longer considered credible. These are steam explosion within the reactor vessel, hydrogen deflagration, and steam explosions in containment. The fourth potential cause of early containment breach, a preexisting opening in containment, is important in severe accidents.

Because containment strengths were previously underestimated, times to containment breach were unrealistically short. Longer times to breach allow a greater degree of aerosol depletion. The development of distributed containment breaching (cracking) limits pressure rise and is likely to prevent gross containment failure while allowing only small quantities of radioactivity to leak out.

The penetration of the base mat by hot core debris has a much lower potential than estimated in WASH-1400 and would lead to only minor releases if it were to occur. Therefore, this mechanism for fission product escape is unimportant.

Pressure suppression pools in boiling-water reactors are very effective in removing fission products, as shown experimentally. The effectiveness of ice condensers in removing fission products is predicted to be high but has not been shown experimentally. In some plants, containment bypass accidents will release fission products under water in flooded compartments; in these cases, the release will be very small. In other plants, the release will take place inside containment as a containment-protected small-breach accident; in these plants, the release also will be very low. In yet other plants, the release will take place in a large auxiliary building; in these cases, it will be necessary to consider the effects of aerosol depletion and other processes, which can be substantial. These considerations point up the importance of plant design differences in calculations of source terms.

The technology of thermal hydraulics is advanced and tested sufficiently for use in accident analyses. The following factors, which may have significant release-attenuating effects, need to be evaluated in depth to ascertain whether further work is necessary: diffusiophoresis, suspended liquid water in containment, compartments within containment, and adjacent buildings and structures surrounding containment.

Revaporization of deposited fission products in the reactor coolant system can occur but is not yet well characterized. However, analyses are generally conservative; some retention in the reactor coolant system will be long-lasting and result in decreases in source terms.

Although several available models differ in predictions of the escape of fission products from overheated fuel, these differences are not significant. Computer programs have undergone substantial independent development since the publication of WASH-1400. Such independent development of codes has provided a certain degree of verification testing. In addition, the data base supporting the computer programs is substantially larger and more complete than that which existed a decade ago. Finally, in the committee's view, many of the difficulties attributed to computer programs are more closely related to differing assumptions and boundary conditions than to the algorithms used. The existing codes are generally conservative representations of the processes involved in severe accidents.

The committee suggested several areas for additional investigation to confirm existing source-term calculations and assure that no important phenomenon has been neglected. The areas should be evaluated in depth to ascertain whether further work is necessary. The committee believes the results of such investigations will not have a major impact on its assessment. These areas include:

- Chemical interactions of fission products with surfaces,
- Tellurium behavior,
- Role of boron in fission product chemistry,
- Rate and mode of core damage progression,
- Role of control rod material,
- Role of contiguous buildings and containment compartments,
- Effect of suspended water and diffusiophoresis,
- Revaporization of deposited fission products,
- Development of a *de minimum* criterion, and
- Uncertainty estimates.

4.1.2.9 EPRI Source-Term Program

The Electric Power Research Institute (EPRI) has undertaken a study to update the accident consequence portions of WASH-1400 and develop more-realistic source terms.²⁹ The report of this work was originally scheduled for completion in late 1983 or early 1984 but has not yet been made publicly available.²⁹ The bulk of EPRI's work on severe-accident technology is funded as part of EPRI's normal programs. Part of this work, however, has been supported through the Industry Degraded Core Rulemaking (IDCOR) program because it is aimed at developing specific information for industry use in the NRC severe-accident rulemaking proceedings (see Sec. 4.1.2.7 for further discussion of the IDCOR source-term project).⁹⁰ The EPRI work for IDCOR has consisted primarily of (1) an analysis of fission product release from a degraded core, (2) development of computer codes to describe core heat-up and degradation for both pressurized- and boiling-water reactors, and (3) analysis of hydrogen distribution and combustion in reactor containment buildings.⁹¹ Additional work performed by EPRI for IDCOR has included the development of a model for Zircalloy oxidation in hydrogen-rich steam environments and an analysis of reactor vessel bottom-head failure resulting from thermal attack by molten core debris.⁹¹ Also, EPRI has contracted with Science Applications Inc. to update accident consequence portions of WASH-1400 and in particular to reanalyze the Surry pressurized-water reactor; this work is in progress.^{92,93} Light-water-reactor safety research and source-term research at EPRI have been described in the literature; the experimental work is described in more detail in Sec. 4.2.^{94,95}

4.2 EXPERIMENTAL STUDIES IN SUPPORT OF SOURCE-TERM REEVALUATION

Many countries have undertaken a wide range of experimental studies in source-term reevaluation.¹⁶ These studies generally seek better data for input to the models of fission product release after severe accidents. They range in size from relatively small laboratory-scale experiments for determining the physicochemical characteristics of fission product releases to large blowdown facilities and reactor loops, several of which are internationally sponsored.¹⁴ Summary information on experimental programs may be found in Tables 10 and 11. Further summary information on experimental programs is tabulated in Ref. 42.

Source-term research projects fall into two broad categories: individual laboratory-scale tests of, for example, the effects of variables such as melting on fission product release or the reactions of fission products with other reactor materials such as metal or water, and large-scale integral experiments of the behavior of fission products as they are transported through a reactor's internal systems under realistic accident conditions. Data from the integral experiments are used to validate the improved models of physical and chemical processes that result from the laboratory-scale work -- models that, in turn, can yield more-realistic source-term estimates.¹³

To better understand the physical and chemical mechanisms of fuel melting, experimental studies in the United States, Great Britain, and West Germany focus on the reactions that can occur in fuel melting, the rates of those reactions, and associated thermal effects. Of particular interest is the extent to which the uranium dioxide fuel is exposed to steam and hydrogen and how this exposure influences the rate of fission product release and the physical conditions of the core.¹³

Laboratory experiments are under way to study the actual release of fission products after overheating has ruptured the zirconium fuel cladding. These experiments seek to precisely identify fission product species, as well as the release rates and physical forms of the released material. Accurate determination of the characteristic fission product release at the point of emission from overheated or melting fuel is complicated by the necessity of ensuring that the experimental sampling system does not itself affect the form of the fission product release. Some experiments are conducted with nonradioactive simulants of fission products. Other experiments are conducted with real fission products, generated either by damaged fuel pins in a reactor loop or by heating samples of highly irradiated fuel in a furnace.^{13,14}

Examples of techniques being used for out-of-reactor laboratory experiments are provided by work at Argonne and Oak Ridge National Laboratories. At Argonne, the fission product emission from a small hole in an inductively heated sample of irradiated spent fuel rods is drawn directly into a mass spectrometer, which identifies the chemical species of the fission products. Initial experiments have investigated fission product release into a vacuum; later experiments are to investigate releases into different mixtures of steam, hydrogen, and air to simulate situations likely to be encountered in failed reactor cores.^{13,14,29} At Oak Ridge, the volatilization rates of fission products and structural materials have been examined by heating kilogram quantities of irradiated fuel. Irradiated fuel samples are melted in an induction furnace and the released material is swept into a thermal gradient tube, in which the temperature drops from

TABLE 10 Topical Summary of Experimental Programs Supporting Source-Term Reassessment^a

Topic	Description	Sponsor	Research Institution
In-vessel release	Fission product release from fuel	EPRI	Argonne National Laboratory
	Fission product release from fuel	NRC	Oak Ridge National Laboratory
	Core melt release	NRC	Oak Ridge National Laboratory
	In-reactor fission product release and transport (Power Burst Facility)	NRC	Idaho National Engineering Laboratory
	In-reactor fission product release and transport (TREAT facility)	EPRI	Argonne National Laboratory
Primary-system thermal hydraulics	In-reactor experiments (Power Burst Facility)	NRC	Idaho National Engineering Laboratory
	Experiments on upper plenum of reactor vessel	EPRI	Westinghouse
	Fuel debris formation and relocation	NRC	Sandia National Laboratories
	Coolant boil-away and damage progression	NRC	Battelle Pacific Northwest Laboratories
	Aerosol transport	International Consortium	Marviken Power Plant, Sweden
Primary-system transport	High-temperature chemistry	NRC	Sandia National Laboratories
	TRAP-MELT code verification	NRC	Oak Ridge National Laboratory
	Interaction between fission products and aerosols	NRC	Oak Ridge National Laboratory
	In-reactor fission product release and transport (Power Burst Facility)	NRC	Idaho National Engineering Laboratory
	Aerosol transport	International Consortium	Marviken Power Plant, Sweden
	Aerosol transport	EPRI	Hanford Engineering Development Laboratory
	Suppression pool experiments	EPRI	Battelle Columbus Laboratory

TABLE 10 (Cont'd)

Topic	Description	Sponsor	Research Institution
Release outside reactor vessel	Interaction of core debris with concrete: system pressure injection and high-pressure injection	NRC	Sandia National Laboratories
Containment transport	Post-accident chemistry (iodine aqueous chemistry)	NRC	Oak Ridge National Laboratory
	Containment system transport	EPRI/NRC	Hanford Engineering Development Laboratories
	Hydrogen containment (FITS facility)	NRC	Sandia National Laboratories
	Containment aerosol transport (Nuclear Safety Pilot Plant Facility)	NRC	Oak Ridge National Laboratory
	Fission product and aerosol ice-scrubbing experiments (Westinghouse facility) (planned)	NRC	Battelle Pacific Northwest Laboratories

^aThis partial list emphasizes experimental work in progress in the United States and is adapted from Ref. 73.

TABLE 11 Summary of Source-Term and Related Reactor Safety Experimental Programs^a (Institutional Listing)

Country and Description	Program Description	Status and Results
United States		
Argonne National Laboratory	Mass spectrometric studies of chemical form of volatilized fission product simulants heated in Knudsen cell. ^{b,c,d}	In progress
Argonne National Laboratory	Study of heat transfer between molten fuel and steam.	In progress
Argonne National Laboratory-West	In-reactor experiments at test reactor TREAT on fission product release from preirradiated fuel rods and physicochemical properties of released fission products. ^{c,d}	Initiated in 1982, due to be completed in 1985
Battelle Columbus Laboratory	Suppression-pool scrubbing measurements for EPRI. ^b	In progress; early results available
EPRI Nuclear Safety Analysis Center	Experiments on internal containment loading from overpressurization and mechanisms of structural failure. ^c	Set of experiments completed just prior to July 1983
	Large-scale tests on formation, combustion, and management of hydrogen in the containment. ^{c,e}	Planned as of July 1983
General Electric Company	Study of suppression pool scrubbing. ^b	In progress; extensive results available
Idaho National Engineering	Power Burst Facility: First integral experiments involving in-reactor tests to core melt conditions with highly instrumented sampling system, measurement of fission product release. Multi-national cooperative program. ^{b,c}	In progress; first test results available, extensive test series planned

TABLE 11 (Cont'd)

Country and Description	Program Description	Status and Results
Idaho National Engineering Laboratory	Loss of Fluid Test (LOFT) reactor experiments sponsored by a number of member countries of the Organization for Economic Cooperation and Development's Nuclear Energy Agency. ^d	In progress
Oak Ridge National Laboratory	CRI-II Facility: Measurement of properties of aerosols released from degraded core heated in steam (4.5-m ³ steel vessel). Laboratory scale (out-of-reactor). ^b	In progress; early results
	Measurement of release rates and chemical characterization of fission products released from high- and low-burnup light-water-reactor fuel. ^{b,c,d,f}	In progress; early results
	Measurement of transmission of aerosols through large-diameter pipes. ^b	In progress
	NSPP Facility: Measurement of aerosol depletion in 38.3-m ³ steel vessel with steam. ^b	Operating for several years; results available
Pacific Northwest Laboratory	Containment Systems Experiment: Experimental data base for aerosol behavior in large (850 m ³) facility. ^b	Previously used for validation of aerosol codes; new experimental program planned
Portland Cement Association	Tensile tests to measure cracking and leakage through containment walls. ^c	Facility constructed

TABLE 11 (Cont'd)

Country and Description	Program Description	Status and Results
Rockwell International	Experiment on fallout behavior of high-temperature, high-concentration uranium dioxide aerosols. ^f	Two relaxation times; rapid agglomeration
Sandia National Laboratories	High-Temperature Fission Product Chemistry and Transport Program: Determination of interactions among fission product and reactor materials and quantitative descriptions of chemistry for inclusion in TRAP-MELT code. ^b	In progress; cesium and tellurium results available
	Estimation of aerosol components of source terms from molten core-concrete interaction. ^b	In progress; preliminary use in BMI-2104 (draft).
	Overpressurization tests on scale-model containment buildings. ^{c,d}	In progress. Tests of electrical penetrations scheduled to begin in January 1984, tests of seals and gaskets underway, tests of mechanical penetrations to begin late 1984.
Canada		
Chalk River	NRU Facility: Measurement of in-reactor release of fission products and aerosols from overheated fuel. ^{b,d}	Proposed start late 1983
Whiteshell	Assessment of fission product chemistry in containments; laboratory-scale. ^b	In progress; early results available

TABLE 11 (Cont'd)

Country and Description	Program Description	Status and Results
England		
Winfrith Heath	Experiments aimed at determining the behavior of multipin fuel clusters under increasingly severe conditions at the Steam Generating Heavy Water Reactor. ^d	In progress
France		
Commissariat a l'Energie Atomique	Structural tests of containment. ^c	In progress
Commissariat a l'Energie Atomique at Grenoble	FLASH Test Program: Measurement of in-reactor release of fission products from preirradiated damaged fuel pins. ^{b,d}	Planned start mid-1983; due to be completed in 1984
Cadarache	Experiment aimed at determining the behavior of multipin fuel cluster under increasingly severe conditions at the PHEBUS reactor. ^d	In progress
Cadarache and Saclay	PITEAS and DENT Facilities: Small-scale (3-m ³ and 6-m ³) validation of aerosol codes with heated floor and cooled walls. ^f	PITEAS planned; DENT under way
West Germany		
Karlsruhe Nuclear Research	SASCHA Facility: Out-of-reactor heating of light-water reactor fuel simulants and measurement of fission product releases. ^{b,c,d,e,g}	Operating since mid-1970s

TABLE 11 (Cont'd)

Country and Description	Program Description	Status and Results
Battelle-Frankfurt	DEMONA Facility: Demonstration of containment response and aerosol behavior in a simulated large-scale (642-m ³) pressurized-water-reactor concrete containment. ^{b,c,d}	Planned start in 1984
Japan		
Central Research Institute for the Electric Power Industry (CRIEPI)	Testing and analysis of containment structural integrity under thermal and pressure loading. ^c	In progress
Sweden		
Marviken	Multinational cooperative program to perform aerosol transport tests: full-scale simulation of aerosol formation and transport in reactor coolant systems for specific sequences. ^{b,c,d,h}	Preliminary tests in April 1983; extensive test series planned

^aThis partial list emphasizes recent major experimental programs.^eRef. 96.^bRef. 29.^fRef. 26.^cRef. 13.^gRef. 55.^dRef. 14.^hRef. 97.

1000°C to 150°C, and from there into a series of high-efficiency filters.⁹⁸ Analysis of the activity deposited in the thermal gradient tube and on the filters provides information on the quantity and chemical form of the released material.^{2,13,14,26,29} A somewhat different technique is being applied at the Kernforschungszentrum Karlsruhe nuclear research center in West Germany. Great Britain is also conducting laboratory work in this area.^{13,14,29,55,96}

Test-reactor programs to study fission product release involve integral experiments that simulate the time and temperature history of specific accident scenarios. Although this type of research is more expensive to conduct than laboratory-scale tests, in-reactor experiments can more accurately define the technical parameters of fission product release. With in-reactor experiments, the difficulties of measuring fission product release are compounded by the necessity of safely operating a reactor facility. Experimental programs tend to evolve cautiously from studies of fuel behavior under increasingly severe conditions to the point of fuel failure and release. Some of the programs go beyond design-basis accident conditions to look at releases that might result from accidents with severe core damage.¹⁴

A program of experiments has been in progress at the research center of the French Commissariat à l'Energie Atomique at Grenoble with single-pin rigs in the Siloe research reactor. The program, known as Flash Combustible Severement Endommage, is examining releases from preirradiated damaged fuel pins in the temperature range from 1350-1800°C -- conditions that might result from a small loss-of-coolant accident. This series of experiments is due to be completed in 1984.^{13,14,29}

In the United States, in-reactor experiments are being performed in Idaho at the Power Burst Facility under NRC sponsorship and at Argonne's TREAT facility, which is sponsored by a consortium that is organized by EPRI and includes Ontario Hydro, DOE, NRC, and Belgonucleaire.^{13,29,94,95}

Among the more important in-reactor experiments directed specifically at fission-product release fractions, rather than a wider range of severe accident phenomena, are Argonne's in-reactor source-term experiments. The experiments use preirradiated fuel rods from the BR3 reactor (a small experimental pressurized-water reactor in Belgium) in Argonne's TREAT facility. The experimental rig incorporates detectors to measure the physicochemical properties of the released fission products as close to the source of emission as possible. This program, which was initiated in 1982, is due to be completed in 1985.^{14,94,95}

Larger and more-dramatic experiments to determine the behavior of multipin fuel clusters under increasingly severe conditions are being carried out in such special-purpose test reactors as the Power Burst Facility in Idaho and the PHEBUS reactor at Cadarache in France. Similar experiments are being conducted in loops at the NRU reactor in Chalk River, Canada, and at the Steam Generating Heavy Water Reactor at Winfrith Heath in the United Kingdom.^{14,29} Although the main purpose of these experiments is to determine coolability under blowdown conditions and the possible consequences if the bundles develop cladding ballooning, they will examine fission product releases in their later stages. The results should be useful for analyses of source terms. The same is true of recent experiments being sponsored at the Loss of Fluid Test

(LOFT) reactor by a number of member countries of the Organization for Economic Cooperation and Development's Nuclear Energy Agency.¹⁴

Aerosol behavior was first studied in connection with sodium-cooled fast reactors, but more recently attention has turned to aerosols that might be expected from severely degraded light-water reactor cores.⁹⁹⁻¹⁰² Results of laboratory-scale experiments with nonradioactive aerosols, especially from the Kernforschungszentrum Karlsruhe research center in Germany, have been used to improve computer codes for predicting aerosol transport. Particularly in moist atmospheres, aerosols can agglomerate readily to form larger particles that can settle out on exposed surfaces or clog up in escape pathways.^{14,27}

Similar aerosol studies are being carried out in other countries. The main problems are development of (1) larger aerosol generators that can produce realistic non-radioactive simulations of the core material mixtures expected from severe accidents and (2) large chambers, simulated structures, and filtering systems to model escape pathways. In France, previous work on aerosols in a dry atmosphere is being supplemented with a series of experiments known as the Piteas project.¹⁰³ These experiments use a 3-m³ stainless steel chamber to study the behavior of aerosols in a wet atmosphere.^{14,29}

Two large experimental projects with international support are being developed to assess the transport and retention of fission products in aerosol form. The first of these, an investigation of aerosol transport in the primary circuit of light-water reactors, is the Marviken project, located at the Marviken power plant in Sweden and supported by organizations from Sweden, Canada, Finland, France, Italy, Japan, the Netherlands, the United Kingdom, and the United States.^{13,14,29,55,97}

The tests, which are approximately full-scale, are to be performed by Studsvik Energiteknik Ab in a decommissioned reactor. The facilities at Marviken, which were once intended for a heavy-water reactor with a large pressure vessel, have already been used as a blowdown test facility. Now a full-scale model pressure vessel is being fitted inside the former reactor pressure vessel, together with a large aerosol generator and other simulated components such as core internals, a pressurizer, and relief chambers. High-concentration aerosols, with densities exceeding 100 g/m³, are to be studied under simulated accident scenarios. The aerosol generator is to be capable of producing more than 100 kg of radioactive material simulating the aerosols that might be expected from a severely degraded reactor core. The \$8.5 million program, expected to continue into 1985, is the first large-scale simulation of fission product transport through a reactor's internal systems. Six large-scale tests are planned to measure aerosol behavior under different conditions.^{13,14,97}

A distinct set of issues must be considered if fission products released from the fuel escape from the primary cooling system and into the containment building. The containment structure's role in determining the accident source term depends partly on the behavior of fuel and fission product aerosols as they are released into the containment.¹³ Another large international project, currently supported by Germany and Switzerland, is known as DEMONA (an acronym for demonstration of nuclear aerosol behavior) and involves the study of aerosol release into a containment building. A one-

quarter-scale model of the containment of the Biblis B reactor at the Battelle-Frankfurt research center, which has previously been used for blowdown experiments as part of the German reactor safety studies, is to be used for the new series of aerosol tests, which are due to be completed by 1985. Although the containment configuration is representative of the German pressurized-water reactors, the project's goal is to demonstrate aerosol depletion in a large-scale facility and further validate computer codes that should have more-general application.

Another issue involves the ability of safety-related equipment in the containment to continue functioning in a high-radiation environment. The EPRI is sponsoring research on containment equipment qualification, and results from relevant source-term research will be incorporated as data become available.^{13,94,95}

Because the magnitude and nuclide distribution of radionuclide release in a severe core accident depend very strongly on the time interval between release of the nuclides from the core and failure of the containment building, source-term reassessment studies are also very concerned with the results of research programs on containment integrity.¹⁰⁴ The WASH-1400 report assumed early catastrophic failure of the containment in a severe accident and, as a result, very little attenuation of fission product release.^{13,14}

The length of time required for a containment structure to fail as a result of overpressure and the manner in which it would fail are subjects of extensive investigation. Recent estimates of containment design safety margins range from 1.5 to 2.5. If a containment is going to fail, most research indicates failure would result from overpressure developing over several days. However, pressure spikes caused by burning the large amount of hydrogen produced from severe core damage might result in earlier failure. Another question under investigation is whether steam explosions or depressurization accidents of sufficient violence could turn debris into missiles capable of penetrating the containment.¹³

These questions are being addressed in a wide range of experimental programs ranging from laboratory-scale tests on the melting characteristics of fuel and associated hydrogen production to destructive pressure tests on model containment structures. Some experiments drop sizeable masses of molten core material into pools of water or onto different types of concrete. Other tests use steam rocket launchers to fire pieces of steel pipe at concrete walls.^{14,55}

The EPRI is investigating both the potential internal containment loading from overpressure and the mechanisms by which the structure could fail. The EPRI Nuclear Safety Analysis Center has recently completed a comprehensive set of experiments and plans large-scale follow-up tests on the formation, combustion, and management of hydrogen in the containment. The work indicates that the effects of hydrogen combustion can be significantly curtailed by the use of igniters and water sprays. Data from these and other experiments are being used to estimate the pressure loading on containment structures.^{13,94,95}

Other EPRI research is being conducted to determine whether overpressure would cause slow leakage rather than catastrophic containment failure. Tensile tests to

measure the cracking and leakage through containment walls are being sponsored by EPRI at Skokie, Illinois, where the Portland Cement Association has constructed a special facility to apply the enormous loads required to crack 4-ft-thick (1.2-m) concrete panels.^{13,94,95}

Complementary overpressure experiments on scale-model containment buildings are being performed by Sandia National Laboratories under NRC sponsorship. These experiments are designed to determine the behavior of containment buildings, including electrical and mechanical penetrations, when subjected to some of the more severe accidents. They include scale-model tests of free-standing steel containments and the design of a reinforced-concrete containment model. Tests of seals and gaskets have been under way while tests of complete electrical penetrations and tests of mechanical penetrations have been scheduled to begin in the early and latter part of 1984 respectively. The program will provide benchmark data for the evaluation of computer codes and provide estimates for leaking rates through penetrations.^{13,14}

Other extensive testing and analysis of containment structural integrity under thermal and pressure loading is being conducted in Japan by the Central Research Institute for the Electric Power Industry.¹³ France is testing containment structures under the direction of the Commissariat à l'Energie Atomique.¹³ In the United States, DOE has defined a comprehensive plan, contingent on congressional funding, to research containment integrity under degraded core conditions.¹³

5 AREAS OF CONTROVERSY AND DIFFERENCES OF OPINION

Areas of controversy and disagreement among different organizations participating in source-term reassessment have become more well defined and limited during the past year. Differences between NRC contractors and IDCOR in most areas seem to be slight although specific areas of disagreement exist.

The NRC and IDCOR continue to be separated by some issues of fission product chemistry, most notably the timing of tellurium release, and containment failure modes, with the NRC unwilling to agree that early, catastrophic containment failures from steam or hydrogen explosions are impossible.¹⁰⁵ However, the two sides agree that source terms can be reduced from the WASH-1400 estimates, but not by how much.

The NRC contractors continue to object that IDCOR analyses underestimate the consequences of a few very serious sequences (especially the boiling-water-reactor Mark I suppression-pool-bypass sequence and the pressurized-water-reactor V sequence in which high-pressure pipes break through valves into low-pressure pipes and fission products can bypass containment), though they agree that the probability of these sequences is low.

The IDCOR appears to conclude that no fatalities will result from any accident sequence (provided that people are evacuated in the 10-mile plume-exposure-pathway emergency-planning zone and sheltered within the 25-mile ingestion-pathway emergency-planning zone). However, the NRC so far appears unwilling to agree that all risk of fatalities from radiation injuries can be discounted even with current evacuation plans.¹⁰⁵

The time scale of severe nuclear accidents will have a major effect on the magnitude of the source terms; this has recently been a topic of controversy. Attempts to determine containment pressure limits have come up against analytical limits; furthermore, inadequacies in state-of-the-art concrete analysis have been reported.⁴⁰ There remains some disagreement about which, if any, mechanisms for early containment failure are possible.¹²

The following areas also warrant additional investigation: rate and mode of core damage;²³ effect of suspended water;²³ diffusiophoresis;²³ role of high-energy photons in changing fission product chemistry;⁴⁰ potential for resuspension of aerosols late in an accident sequence;⁴⁰ core-concrete interactions;⁴⁰ uncertainty about basic thermal hydraulic behavior (including modeling changes in core geometry as a melt progresses);⁴⁰ and uncertainty about recirculation flows.⁴⁰

The behavior of aerosols and their effects on the magnitude of source-term values has been an area of controversy.³² Another area of uncertainty is natural circulation from the core to the plenum, which reportedly might increase fission product retention by a factor of 100 over earlier expectations.⁴⁰ The possibility of decay heat reevaporizing fission products that plate out on internal surfaces of the system is being widely studied and warrants additional investigation.²³

Fission product chemistry, particularly the roles of tellurium and boron, remains controversial.²³ (Some additional points of controversy about fission product chemistry are discussed in Sec. 3.2 in connection with individual radionuclide categories.)¹⁰⁶ The controversy over fission product chemistry is due in part to the difficulty of predicting coolant water impurities and their effects on chemical reactions.⁴⁰ Related questions concern suppression-pool scrubbing; if accumulations of fission products in a pool act as surfactants, the pool's ability to scrub fission products as an accident progresses could be significantly reduced.⁴⁰ Some uncertainty remains about whether water causes scrubbing when it contacts melt material.⁴⁰

Although results of IDCOR and NRC studies on hydrogen production, burning, and explosions agree fairly well, it has been reported that NRC representatives were concerned that IDCOR may have treated hydrogen production inadequately.⁴¹ These NRC representatives also questioned IDCOR's modeling of heat up in the upper plenum and hydrogen production during actual core melting.⁵⁰ More work is needed to determine how flames from a burn would spread among compartments and in dense aerosols.

Significant disagreement exists over steam explosions.⁵⁰ While IDCOR researchers have looked at the energy necessary to substantially deform or rupture a containment and concluded that enough material could not accumulate to generate an explosion large enough to either threaten the reactor pressure vessel or the containment, NRC and national laboratory representatives disagree. Representatives of industry, NRC, and national laboratories also disagree about the computer codes in use by the different participants and the extent to which different phenomena are suitably incorporated in or missing from the codes.^{40,50} Section 4.1.2.8 discusses some additional areas where further investigation is desirable.

Specifically expressed areas of controversy and differences of opinion between the American Physical Society study group and IDCOR or NRC staff and contractors have not yet officially emerged, because the study group has not yet reported its results (as of September 30, 1984). However, there have been long-standing indications of some concern among study group members that our understanding of severe-accident phenomena remains uncertain in too many respects, and that, for this reason, more source-term research is needed. A concern that the research results should be fully accessible to the scientific community has been expressed more recently.

6 RECENT DEVELOPMENTS IN SOURCE-TERM REEVALUATION

The most recent general meeting on source-term issues held before the cutoff date of this report, September 30, 1984, was the American Nuclear Society Topical Meeting on Fission Product Behavior and Source Term Research held in Snowbird, Utah, July 16-19, 1984.¹⁰⁷ This meeting was held to discuss the most recent information on source terms and to examine whether a sufficient consensus on source terms and related topics was at hand to provide for a change in regulation requirements.²⁵

No overall consensus on closure was reached at the meeting. Some participants suggested that enough was already known for the NRC to rule on new lower source terms and to reduce emergency planning requirements, while other participants were more cautious, maintaining that some important questions remain unanswered.⁷⁰

Among the participants, however, there was a consensus that WASH-1400 source terms incorporated large factors of conservatism. Most participants agreed that WASH-1400 release values are in most instances an upper bound. Fission product release estimates for many accident sequences were reported by various participants to be lower by factors of 10 to 100 to even 1000, but a few estimates stayed high and a very few were higher than WASH-1400 values.⁷⁰ However, the meeting produced less of a consensus on the extent to which source-terms could be reduced at this time.²⁵

The unevenness of research results, with fission product releases varying by reactor type and accident sequence, has led to a broadening of NRC's source-term focus in two major areas. First, the NRC staff has expanded its focus to include the evaluation of the statistical probability of each severe accident sequence and tentatively proposed ignoring accidents with probabilities lower than an established limit (possibly one per billion reactor years), rather than having source terms dictated by worst-case accidents regardless of their probabilities of occurrence. Second, the focus has been expanded to include containment loading and containment bypass studies to evaluate probabilities of breaches of each containment type, since the most serious accidents all involve bypass or early breach of containment.⁷⁰

Although a few accidents studied still involved serious releases, the highest-release accident sequence studied for pressurized-water reactors (the V sequence, in which a break occurs outside containment) was described as a plant-specific problem for which releases now appear much lower due to water scrubbing.^{70,85}

Uncertainties remain about fission product behavior and chemical forms; however, treating all iodine as cesium iodide and all cesium as cesium hydroxide was a widely used approximation that worked well.⁷⁰

Research papers on the whole supported the NRC position that tellurium would be released from a damaged core later than iodine and cesium. (Researchers in the IDCOR program estimated that tellurium would be released sooner; thus IDCOR calculations of the amount of fission products available for release at specific points as an accident sequence progresses differed from those of other researchers.⁸⁵) Greater tellurium release was reported to occur with more steam in the system. Tellurium is

chemically held by zirconium, bonds with iron and nickel surfaces, and plates out on piping.⁷⁰

Recent results on fission product behavior during the accident at Three Mile Island 2 were also reported at the meeting. The first results from DOE-sponsored tests of Three Mile Island's damaged fuel and measurements in the reactor indicate that some important parts of computer codes used to model severe accidents might need further work.⁷⁰

In the accident, significant melting of structures and fuel cladding took place earlier than previously believed. Evidence was presented that the fuel was damaged in a different manner than had previously been thought; collapsing fuel blocked steam from traveling upward and forced decay heat back into the core. Core damage appeared to be extensive, with no more than two fuel assemblies remaining intact. The fuel rods from the periphery collapsed toward the center. All computer codes underestimated the actual extent of core damage.⁷⁰

Fuel samples from Three Mile Island 2 showed that uranium had dissolved in melted zirconium cladding. Cladding-water reaction, which produces hydrogen, proceeded at a much slower rate than current codes anticipate (apparently in part because operators turned off pumps that reduced the amount of steam available during the accident).⁷⁰

Several papers reported on the fission products produced in the accident and where they were transported and deposited. Locations where fission products were sampled included the upper plenum, reactor coolant system, and basement sludge. Codes developed by both NRC and industry predicted iodine and cesium would be produced at about the same rate, but less than half as much iodine was located in the Three-Mile-Island 2 sampling as cesium; more radioactive cesium was located than was predicted to be released by the codes.⁷⁰

The presence of silver plated out in the upper plenum was significant. Personnel from NRC and IDCOR have contended the timing of silver's aerosol formation.⁷⁰ (Aerosol production and timing is of concern because estimates of fission products available for release to the containment depend strongly on what aerosols are available.) The IDCOR computer codes had modeled silver in the control rods as melting harmlessly into the bottom of the plenum at temperatures lower than those at which key fission products are released; however, NRC personnel maintained that the Three-Mile-Island results show that silver must have been available as an aerosol.⁷⁰

Among general results from the meeting, several groups reported that fission product transport was heavily dependent on both the plant materials available for binding and the amount of hydrogen in steam.⁷⁰ Revaporization of fission products from decay heat after initial deposition on reactor coolant system or containment surfaces was a topic of major continuing concern.⁷⁰

Since the American Nuclear Society topical meeting on fission product behavior and source-term research, NRC and IDCOR representatives have met to discuss remaining differences (see Sec. 5). Also, preliminary information on the Society's Special

Committee on Source Terms study results were reported at the Fifth International Conference on Thermal Nuclear Reactor Safety held in Karlsruhe, West Germany, on September 10-13, 1984. However, the final reports of the IDCOR, American Nuclear Society, and American Physical Society studies have not yet been released, and the NRC rulemaking and report on source terms will probably not be released before some or all of these studies are completed.

The *Report of ANS Special Committee on Source Terms*, dated September 1984, was scheduled for release at the American Nuclear Society/European Nuclear Society Conference in Washington on November 12, 1984.⁴² Preliminary information indicates that the committee found that source terms were one or more orders of magnitude lower than those in WASH-1400. Technical progress achieved in both fundamental knowledge and analytical techniques since the benchmark study in 1975 led to the lower estimates.

To indicate the magnitudes of reductions in source terms for a broad spectrum of accidents, a comparison of values for WASH-1400 release category PWR-2 with releases based on the current methods is given in Table 12 for 19 postulated accidents at the Surry reactor, which is considered prototypical of the pressurized-water reactors. For most of the sequences the source terms were reduced to a fraction (0.01% to a few percent) of the estimates in WASH-1400. In the case of the containment bypass accident (V sequence), if the postulated break in the piping of the low-pressure emergency core cooling system occurred at a location submerged in water draining from the refueling water storage tank, the releases could be as small as the values shown in Table 12. Current calculations for the five other pressurized-water reactors studied are similarly lower than the WASH-1400 values.

The releases in the containment bypass accident (V sequence) are very plant-specific. In some plants, containment bypass accidents would release fission products under water in flooded compartments; in these cases, the release would be very small. In other plants, the release would take place inside containment, as a containment-protected small-break loss-of-coolant accident; in these plants the release would also be low. In yet other plants the release would take place in the large auxiliary building; in these cases, aerosol depletion and other depletion processes would be effective in reducing the radioactive release.

The comparisons of values for WASH-1400 release categories BWR-2 and BWR-3 with releases based on current methods are given in Table 13 for several postulated accidents and for each of the two reactors, Shoreham and Peach Bottom, that are considered prototypical of boiling-water reactors. Current calculations are again lower for these two as well as for three other boiling-water reactors studied. Although the releases for boiling-water reactors are higher than those for pressurized-water reactors, according to the American Nuclear Society the high values are attributed to various factors assumed in the calculations for boiling-water reactors.⁴² Releases for the two reactor types would not differ greatly if the following factors are properly accounted for in the calculations:⁴²

- Differences in assumptions (for example, containment opening size);
- Whether aerosol attenuation in the reactor building was taken into account;

TABLE 12 Comparison of WASH-1400 Release Category PWR-2 with Releases Based on Current Methods

Plant and Accident Sequence	Postulated Containment Opening ^a		Study	Percentage of Core Inventory Released to Environment		
	Area (ft ²)	Time (hr)		Iodine	Cesium	Tellurium
PWR-2	--	2.5	WASH-1400	70	50	30
Surry						
AB- β ^b	0.35	--	SWEC	6.4	6.3	3.9
AB- β ^b	0.1	--	SWEC	1.2	1.2	1.2
AB- γ	7.0	4.5	Battelle	5.7	5.9	14
AB- δ ^b	1.0	0.5	SWEC	8.6	8.6	7.3
AB- δ ^b	0.1	0.5	SWEC	1.6	1.6	1.6
AB- δ ^b	1.0	3.0	SWEC	0.86	0.86	0.73
AB- ϵ ^c	--	24	Battelle	0.0048	0.0047	0.0040
AB- δ ^b	1.0	24	SWEC	0.040	0.040	0.19
AB- δ ^b	0.1	24	SWEC	0.030	0.030	0.14
TMLB- β ^b	1.0	--	SWEC	1.5	1.3	6.4
TMLB- β ^b	0.1	--	SWEC	0.21	2.1	2.3
TMLB- δ	7.0	2.5	Battelle	4.6	5.9	11
TMLB- ϵ ^b	1.0	3.0	SWEC	1.5	1.3	6.4
TMLB- ϵ ^c	--	12.0	Battelle	0.28	0.017	8.1
TMLB- δ ^b	1.0	27	SWEC	0.0036	0.0035	0.018
TMLB- δ ^b	0.1	27	SWEC	0.0028	0.0027	0.014
V	0.2	--	Battelle	41 ^d	40	12
V ^e	0.2	--	Battelle	7.9	7.3	2.5
V ^e	0.2	--	SWEC	1.0	1.0	0.65

Source: Adapted from Ref. 42.

^aIf no time is given, opening is preexisting.

^bRetention in reactor coolant system is taken from Ref. 35.

^cBased on base mat melt-through. Retention in earth outside containment is neglected.

^dIn this accident sequence, core inventory will probably be released into containment or into a compartment that will be flooded by the refueling water storage tank. If so, releases to the environment would be much lower.

^eWith flooding to a depth of 3 ft over the postulated pipe break.

TABLE 13 Comparison of WASH-1400 Release Categories BWR-2 and BWR-3 with Releases Based on Current Methods

Plant and Accident Sequence	Postulated Containment Opening		Study	Percentage of Core Inventory Released to Environment		
	Area (ft ²)	Time (hr) ^a		Iodine	Cesium	Tellurium
BWR-2	--	30.0	WASH-1400	90	50	30
BWR-3	--	30.0	WASH-1400	10	10	30
Peach Bottom						
TC	0.2	13.0	IDCOR	3.0	3.0	7.0
TC-γ ^b	7.0	0.97	Battelle	24	21	37
TC-γ ^b	7.0	0.97	Battelle	10	9.1	25
TW	7.0	29.2	Battelle	4.8	4.5	19
TW	0.1	34.0	IDCOR	13	13	13
AE	7.0	0.56 ^c	Battelle	34	33	65
TQUV	--	18.0	IDCOR	5.0	5.0	4.0
Shoreham						
TC	--	0.5	Science Applications	2.3	2.5	0.5
TW	--	24.0	Science Applications	0.02	2.5	0.3
AV	--	1.5	Science Applications	16	2.8	2.5
TQUV	--	--	Science Applications	0.03	0.009	0.004

Source: Adapted from Ref. 42.

^aCertain boiling-water reactor accident sequences lead to containment breach before core degradation. Time shown is from initiating event to containment breach, except where indicated.

^bRelease via standby gas treatment system.

^cContainment breach occurs after core degradation.

- Assumption of excessive suppression-pool bypass in cases with large containment openings;
- An error in one of the scrubbing models used for suppression pools; and
- Use of inappropriate thermal hydraulics modeling.

As noted in the preceding section, however, NRC contractors continue to object to industry analyses underestimate the source terms for a few very serious accident sequences for both pressurized- and boiling-water reactors.¹⁰⁵ In general, the overall results, conclusions, and recommendations in the report of American Nuclear Society Special Committee are similar to those presented in this report.

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APPENDIX A: DESCRIPTION OF REACTORS AND ACCIDENT SEQUENCES STUDIED IN SOURCE-TERM REASSESSMENTS

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2. Gieseke, J.A., *Summary of Source Term Analyses for Five LWR Plants*, Transactions of the Eleventh Water Reactor Safety Research Information Meeting, U.S. Nuclear Regulatory Commission Report NUREG/CP-0047 (1983).
3. Gieseke, J.A., et al., *Radionuclide Release under Specific LWR Accident Conditions: Volume I, PWR, Large, Dry Containment*, Battelle Columbus Laboratories Draft Report BMI-2104, Vol. 1 (July 1983).
4. Gieseke, J.A., et al., *Radionuclide Release under Specific LWR Accident Conditions: Volume II, BWR, Mark I Design*, Battelle Columbus Laboratories Draft Report BMI-2104, Vol. II (July 1984).
5. Gieseke, J.A., et al., *Radionuclide Release under Specific LWR Accident Conditions: Volume III, BWR, Mark III Design*, Battelle Columbus Laboratories Draft Report BMI-2104, Vol. III (July 1984).
6. Gieseke, J.A., et al., *Radionuclide Release under Specific LWR Accident Conditions: Volume IV, PWR, Ice Condenser Containment Design*, Battelle Columbus Laboratories Draft Report BMI-2104, Vol. IV (July 1984).
7. Gieseke, J.A., et al., *Radionuclide Release under Specific LWR Accident Conditions: Volume V, PWR, Large, Dry Containment Design (Surry Plant Recalculations)*, Battelle Columbus Laboratories Draft Report BMI-2104, Vol. V (July 1984).
8. Gieseke, J.A., et al., *Radionuclide Release under Specific LWR Accident Conditions: Volume VI, PWR, Large, Dry Containment Design (Zion Plant)*, Battelle Columbus Laboratories Draft Report BMI-2104, Vol. VI (July 1984).
9. *IDCOR Program Plan*, prepared by Technology for Energy Corporation, Knoxville, Tenn. (Nov. 1981).
10. *Nuclear Power Plants in the United States, January 1, 1984*, Atomic Industrial Forum, Inc., Washington, D.C. (Jan. 1, 1984).
11. *Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants*, U.S. Nuclear Regulatory Commission Report NUREG-75/014 (previously published as WASH-1400) (Oct. 1975).

12. Silver, E.G., *Operating U.S. Power Reactors*, Nuclear Safety, 24(2):251-255 (March-April 1983).
13. *Volume BWR-x, Book x, Plant Descriptions-Histories*, Nuclear Power Experience, Inc. (1976 et seq.).
14. *Volume PWR-1, Book 2, Plant Descriptions-Histories*, Nuclear Power Experience, Inc. (1976 et seq.).
15. *World List of Nuclear Power Plants*, Nuclear News, Vol. 26, No. 2 (Feb. 1983).

TABLE A.1 Reactor Design and General Operating Data

Plant	Location	Utility	Reactor Type	Start-up date (full power date)
Surry	Surry Co., Va.	Virginia Electric Power Co.	Westinghouse pressurized- water reactor; large dry con- tainment	1972 (1973)
Peach Bottom	York Co., Pa.	Philadelphia Electric Co.	General Electric boiling-water reactor; Mark I containment	1974 (1974)
Grand Gulf	Port Gibson, Miss.	Mississippi Power & Light Co.	General Electric boiling-water reactor; Mark III containment	1989
Sequoyah	Daisy, Tenn.	Tennessee Valley Authority	Westinghouse pressurized- water reactor; ice condenser design	1981 (1982)
Zion	Zion, Ill.	Commonwealth Edison Co.	Westinghouse pressurized- water reactor; large dry containment	1973 (1974)

TABLE A.2 Accident Sequences Studied^a

Plant	Study			
	Battelle	IDCOR	EPRI	WASH-1400
Surry	AB, TMLB', S ₂ D, V	Steam generator tube rupture, V	Multiple Sequences	Multiple Sequences
Peach Bottom	TC, AE, TW	TC, TW, TQUV, TQVW, AEG, S ₁ E	-	Multiple Sequences
Grand Gulf	TPI, TQUV, TC	T ₂₃ QW, T ₁ QW, T ₂₃ C, S ₁ , T ₂₃ PQI, T ₁ QV, T ₁ PQI, T ₂₃ PQE, AE	-	-
Sequoyah	TML, TMLB', S ₂ HF	S ₂ D, S ₂ H, V, S ₂ HF, T ₂₃ ML, S ₁ H, S ₁ D, AD, TMLB	-	-
Zion	TMLB', S ₂ D	1, 2, 5, 6, 7, 9, 10, 14, 16, S ₁ CDY, THF	-	-

^aSee Apps. B and C for keys to accident sequence designations.

TABLE A.3 Reactor Core Data

Reactor	Power Level			Active Core Size (m)		Fuel Inventory (t)	Number of Fuel Elements	Pitch (cm)		Power Rating (kW/L)		Expected Burn-up (MW·d/t)
	MW(t)	MW(e) gross	Design Expectation MW(e)	Height	Diameter			Rod	Assembly	Peak	Average	
Surry 1	2441	848	811	3.66	3.03	70	157	1.43	21.5	231.7	92	31,500
Surry 2	2441	824	-	3.66	3.03	70	157	1.43	21.5	231.7	92	31,500
Peach Bottom 2	3293	1098	1098	3.81	4.75	149	764	1.63	15.24	132	49.19	25,000
Peach Bottom 3	3293	1098	1098	3.61	4.75	149	764	1.63	15.24	132	49.19	25,000
Grand Gulf 1	3833	1306	-	3.81	4.8	166.9	800	1.62	15.24	122.2	54.1	28,000
Grand Gulf 2	3833	1306	-	3.81	4.8	166.9	800	1.62	15.24	122.2	54.1	28,000
Sequoyah 1	3411	1183	1221	3.65	3.35	101	193	1.26	21.5	--	103.5	36,000
Sequoyah 2	3411	1183	1221	3.65	3.35	101	193	1.26	21.4	--	103.5	36,000
Zion 1	3250	1085	-	3.64	3.37	87.7	193	1.43	21.5	232	100	33,000
Zion 2	3250	1085	1085	3.64	3.57	87.7	193	1.43	21.5	232	100	33,000

TABLE A.4 Characteristics of Reactor Vessels and Containments

Reactor	Material ^b	Vessel ^a			Containment	Pressure (kg/cm ²)
		Wall Thickness (mm)	Dimensions (m)			
			Height	Diameter		
Surry 1 and 2 ^c	SA 508 Cl 12	203	12.3	3.99	reinforced concrete and stainless steel	3.66
Peach Bottom 2	ASME SA 336	160	22	6.4	pressure suppression	1.68
Peach Bottom 3	ASME 160	6.4	22	6.4	pressure suppression	--
Grand Gulf 1 and 2	Stainless- steel-clad carbon steel	213	22.3	6.38	Mark III	1.05
Sequoyah 1	SA 533	215	13.3	4.4	double	0.84
Sequoyah 2	SA 538	215	13.3	4.4	double	0.84
Zion 1 and 2 ^c	SA 302 B	214	12.6	4.38 i.d.	reinforced concrete and stainless steel	3.3

^aAll reactor vessels are cylindrical.

^bMaterial designations as defined in the *American Society of Mechanical Engineers Boiler and Pressure Vessel Code*, Sec. 2 (1980).

^cCharacteristics of both reactors are identical.

TABLE A.5 Fuel Element Data^a

Reactor	Enrichment (%)	No. Pins/ Element	Linear Fuel Rating (kW/m)	Dimensions (cm)		Maximum Can Temp- erature (°C)	Maximum Fuel Temperature at Core Center (°C)	Cladding	
				Pin	Pellet			Material	Thickness (mm)
Surry 1 and 2 ^b	2.5 avg.	205	50	1.072	0.93 x 1.76	347	2232	Zircalloy 4	0.618
Peach Bottom 2 and 3 ^b	2.99	62	43.96	1.041	1.041	294	2457	Zircalloy 2	0.813
Grand Gulf 1	1.71 avg.	62	19.5	1.227	1.041	293	1891	Zircalloy 2	0.81
Grand Gulf 2	1.71 avg.	62	19.5	1.227	1.041	204	1891	Zircalloy 2	0.81
Sequoyah 1 and 2 ^b	2.1,2.6,3.1	264	41.3	0.95 o.d.	0.85 o.d.	...	1871	Zircalloy 4	0.57
Zion 1	3.0 equivalent	204	21.98	3.64 x 1.07	1.52 x 0.93	347	2313	Zircalloy	0.61
Zion 2	3.0 equivalent	204	21.98	3.64 x 1.07	1.52 x 0.93	347	2343	Zircalloy	0.61

^aAll reactors studied use pellets of uranium oxide (UO₂) for fuel.

^bCharacteristics of both reactors are identical.

TABLE A.6 Reactor Cooling and Moderating Data^a

Reactor	Coolant						Moderator		
	Pressure (kg/cm ²)	Inlet Temperature (°C)	Outlet Temperature (°C)	No. Primary Pumps	Mass flow (t/hr)	No. Loops	Type	Weight (t)	Average Temperature (°C)
Surry 1	158	284	318	3	15,225	3	indirect	--	301
Surry 2	157	284	318	3	15,225	3	indirect	--	301
Peach Bottom 2 and 3 ^b	72	277	-	2	7,720	2	direct	407	saturation
Grand Gulf 1 and 2 ^b	73.1	278	286	2	15,377	4	direct	280	286
Sequoyah 1	158	286	323	3	14,900	4	indirect	131.6	305.2
Sequoyah 2	157.5	285	312	4	14,900	4	indirect	131.6	305
Zion 1 and 2 ^b	157	276.8	312.3	4	15,900	4	indirect	252	295

^aAll reactors are light-water-cooled and moderated.

^bCharacteristics of both reactors are identical.

TABLE A.7 Reactor Control System Data

Reactor	No. of Rods			Material	Other Control Systems	Control Rod Drives
	Coarse	Fine	Safety			
Surry 1	32	5	16	composite of silver, indium, and cadmium	burnable poison, chemical shim	magnetic latch
Surry 2	32	5	16	composite of silver, indium, and cadmium	burnable poison, chemical shim	magnetic jack
Peach Bottom 2 and 3	185	-	-	stainless-steel-clad boron carbide	gadolinium oxide	hydraulic
Grand Gulf 1 and 2	193	-	-	boron carbide	gadolinium oxide	hydraulic
Sequoyah 1	29	8	24	composite of silver, indium, and cadmium	burnable poison, chemical shim	magnetic latch and gravity
Sequoyah 2	29	8	24	composite of silver, indium, and cadmium	burnable poison	magnetic latch and gravity
Zion 1	53 ^a	8	24	composite of silver, indium, and cadmium	boron, chemical shim	magnetic jack
Zion 2	53 ^b	8	24	composite of silver, indium, and cadmium	boron, chemical shim	magnetic jack

^aTwo rods per assembly.

^bTwenty rods per assembly.

APPENDIX B: KEY TO ACCIDENT SEQUENCE DESIGNATION

Accident sequences are designated by a sequence of symbols as illustrated in Fig. B.1. Keys to the interpretation of the accident sequence symbols may be found in the accompanying two tables, Table B.1 for pressurized-water reactors and Table B.2 for boiling-water reactors, respectively. Data for these tables were taken from the following references:

1. Denning, R.S., presentation to American Physical Society Study Group Meeting, Oak Ridge National Laboratory (Dec. 3, 1983).
2. Pasedag, W.F., R.M. Blond, and M.W. Jankowski, *Regulatory Impact of Nuclear Reactor Accident Source Term Assumptions*, U.S. Nuclear Regulatory Commission Report NUREG-0771 (June 1981).

FIGURE B.1 Order of Symbols Used to Designate Accident Sequences

TABLE B.1 Key to Accident Sequence Symbols for Pressurized-Water Reactors

Symbol	Explanation
A	Intermediate to large loss-of-coolant accident.
B	Failure of electric power to engineered safety features.
B'	Failure to recover either on-site or off-site electric power within about 1-3 hr of an initiating transient loss of off-site AC power.
C	Failure of the containment-spray injection system.
D	Failure of the emergency-core-cooling injection system.
F	Failure of the containment-spray recirculation system.
G	Failure of the containment heat-removal system.
H	Failure of the emergency-core-cooling recirculation system.
K	Failure of the reactor protection system.
L	Failure of the secondary steam relief valves and the auxiliary feedwater system.
M	Failure of the secondary steam relief valves and the power conversion system.
Q	Failure of the primary safety relief valves to reclose after opening.
R	Massive rupture of the reactor vessel.
S ₁	A small loss-of-coolant accident with an equivalent pipe-break diameter of about 2 in. to 6 in.
S ₂	A small loss-of-coolant accident with an equivalent pipe-break diameter of about 1/2 in. to 2 in.
T	Transient event.
V	Failure of low-pressure injection system check valve.
α	Containment rupture due to steam explosion in the reactor vessel.
β	Containment failure due to inadequate isolation of containment openings and penetrations.
γ	Containment failure due to burning hydrogen.

TABLE B.1 (Cont'd)

Symbol	Explanation
δ	Containment failure due to overpressure.
ϵ	Containment vessel melt-through.

TABLE B.2 Key Accident Sequence Symbols for Boiling-Water Reactors

Symbol	Explanation
A	Rupture of reactor coolant boundary with an equivalent diameter of greater than 6 in.
B	Failure of electric power to engineered safety features.
C	Failure of the reactor protection system.
D	Failure of vapor suppression.
E	Failure of emergency-core-cooling injection.
F	Functional failure of emergency core cooling.
G	Failure of containment isolation to limit leakage to less than 100 volume percent per day.
H	Failure of core-spray recirculation system.
I	Failure of low-pressure recirculation system.
J	Failure of high-pressure service water system.
M	Failure of safety or relief valves to open.
P	Failure of safety or relief valves to reclose after opening.
Q	Failure of normal feedwater system to provide core make-up water.
S ₁	Small pipe break with an equivalent diameter of about 2 in. to 6 in.
S ₂	Small pipe break with an equivalent diameter of about 1/2 in. to 2 in.
T	Transient event.
U	Failure of high-pressure coolant injection or reactor core isolation cooling to provide core make-up water.
V	Failure of low-pressure emergency-core-cooling system to provide core make-up water.
W	Failure to remove residual core heat.
α	Containment failure due to steam explosion in reactor vessel.
β	Containment failure due to steam explosion in containment.

TABLE B.2 (cont'd)

Symbol	Explanation
γ	Containment failure due to overpressure; release to reactor building.
γ'	Containment failure due to overpressure; release direct to atmosphere.
δ	Containment isolation failure in drywell.
ϵ	Containment isolation failure in wetwell.
z	Containment leakage greater than 2400 volume percent per day.
η	Failure of reactor building isolation.
θ	Failure of standby gas treatment system.

APPENDIX C: DESCRIPTION OF ACCIDENT SEQUENCES STUDIED BY IDCOR

Plant and Accident Sequence	Description
Sequoyah	
S_2D	Small loss-of-coolant accident (S_2) followed by failure of emergency-core-cooling injection system (D).
S_2H	Small loss-of-coolant accident (S_2) followed by failure of emergency-core-cooling recirculation system (H).
V	Failure of a series of two check valves in one of the low-pressure injection coolant systems allows high-pressure primary-system water to enter and rupture the low-pressure piping outside containment (V).
S_2HF	Small loss-of-coolant accident (S_2) followed by failure of the emergency-core-cooling recirculation system (H) and containment-spray recirculation system (F).
T_{23}^{ML}	Transient event (T_{23}) other than loss of off-site power followed by failure of power conversion system (M) and auxiliary feedwater system (L).
$T_1B_3^{MLB'}_{13}$	Loss of off-site power or loss of network load (T_1) followed by failure of diesel generators to provide emergency AC power (B_3). Loss of both the power conversion system (M) and the auxiliary feed-water system (L) is assumed as well as the inability to recover off-site power or emergency AC power within a period of about 1-3 hr (B'_{13}).
S_1H	Small loss-of-coolant accident (S_1) followed by failure of emergency-core-cooling recirculation system (H).
S_1D	Small loss-of-coolant accident (S_1) followed by failure of the emergency-core-cooling injection system (D).
Zion	
1	Small loss-of-coolant accident followed by failure of recirculation cooling.
2	Seismic event followed by loss of all AC power.

APPENDIX C (Cont'd)

Plant and Accident Sequence	Description
Zion (Cont'd)	
5	Loss of main feedwater followed by anticipated transient without shutdown, failure to control pressure rise (i.e., failure of augmented auxiliary feedwater or primary pressure relief).
6	Turbine trip followed by anticipated transient without shutdown, failure to control pressure rise (i.e., failure of augmented auxiliary feedwater or primary pressure relief).
7	Spurious safety injection followed by failure to control the safety injection and recirculation cooling.
9	Large loss-of-coolant accident followed by failure of low-pressure injection.
10	Medium loss-of-coolant accident followed by failure of low-pressure injection.
14	Turbine trip due to loss of off-site power followed by loss of all AC power and failure of auxiliary feedwater.
16	Loss-of-coolant accident in the interfacing system (residual-heat-removal inlet valves).
S ₁ CDY	Small loss-of-coolant accident followed by failure of injection and all containment safeguards.
THF	Spurious safety injection followed by failure to control safety injection. Recirculation unavailable.
Peach Bottom	
TW	Transient event (T) requiring reactor shutdown followed by failure of decay heat removal (W).
TC	Transient event (T) requiring reactor shutdown followed by failure to make reactor subcritical (C).
TQUV	Transient event (T) requiring reactor shutdown and followed by failure of feedwater (Q), high-pressure coolant injection and reactor core isolation cooling (U), and low-pressure emergency-core-cooling systems (V).

APPENDIX C (Cont'd)

Plant and Accident Sequence	Description
Peach Bottom (Cont'd)	
S_1E	Small pipe break (S_1) followed by the failure of the emergency-core-cooling injection system (E).
TQVW	Loss of all AC power followed by failure of all emergency systems.
AEG	Large loss-of-coolant accident followed by failure of injection systems and containment.
Grand Gulf	
$T_{23}QW$	Transient event other than loss of off-site power followed by the failure of the power conversion system (Q) and residual-heat-removal system (W) to remove decay heat from containment within 30 hr.
T_1QW	Loss of off-site power (T_1) followed by failure of the power conversion system (Q) and failure of the residual-heat-removal system (W) to remove decay heat from the containment within 30 hr.
$T_{23}C$	Transient event other than loss of off-site power (T_{23}) followed by a failure to make reactor subcritical (C).
SI	Small loss-of-coolant accident (S) followed by a failure of the residual-heat-removal system to remove decay heat from the suppression pool (I).
$T_{23}PQI$	Transient event other than loss of off-site power (T_{23}) followed by failure of a safety or relief valve to reseal (P), failure of the power conversion system (Q), and failure of the residual-heat-removal system to remove decay heat from the suppression pool within 28 hr (I).
T_1QUV	Loss of off-site power (T_1) followed by failure of the power conversion system (Q) and failure of the high-pressure (U) and low-pressure injection systems (V) to provide emergency core cooling.
T_1PQI	Loss of off-site power (T_1) followed by failure of a safety or relief valve to reseal (P), failure of the power conversion system (Q), and failure of the residual-heat removal system to remove decay heat from the suppression pool within 28 hr (I).

APPENDIX C (Cont'd)

Plant and Accident Sequence	Description
<hr/>	
Grand Gulf (Cont'd)	
T ₂₃ ^{PQE}	Transient event other than loss of off-site power (T ₂₃) followed by failure of a safety or relief valve to reseat (P), failure of power conversion system (Q), and failure of the emergency-core-cooling system to maintain coolant inventory (E).
AE	Large loss-of-coolant accident (A) followed by failure of emergency-core-cooling system to maintain coolant inventory (E).

APPENDIX D: NARRATIVE DESCRIPTIONS OF SELECTED ACCIDENT SEQUENCES

The accident sequences described below cover most of the spectrum of serious accidents. Most accidents would have less-severe consequences than those selected for comparison here. The descriptions below are adapted from information supplied by W. Stratton of the American Nuclear Society in LaGrange Park, Ill. (1983).

Pressurized-Water Reactor: Large-Break Loss-of-Coolant Accident with Failure of Core Makeup (AB)

This sequence would result in the loss of primary system water. When core makeup subsequently fails, the core would boil dry. System pressures would be low after the initial blowdown. When the water level reaches about one-third the core height, significant oxidation of the zircalloy cladding and major failure of the core would be expected. Fission products would be swept from the fuel by a mixture of steam and hydrogen that results from the oxidation of zirconium. The vaporizing and recondensing of control rod constituents would produce aerosols. The steam flow would pass at fairly low velocity over the upper internal structures and aerosols would plate out. If these upper internal surfaces reheat after the steam flow ceases, some plated materials could revolatilize. If the break occurs in the coolant inlet pipe, the discharge flow would pass through the steam generator until the lower core was uncovered and a shorter flow path became available through the downcomer.

Previously deposited materials would not reentrain unless a great deal of steam was produced when the core falls into the water remaining in the bottom head. Some dislodgement of fission products might be expected from surfaces in the upper internal structures, but, if the break is a cold leg rupture, dislodgement from the steam generators should be slight because the shorter steam path to the break would be through the cold-leg flow path.

Containment sprays and fan coolers would work during the accident, so debris would be expected to contact water after leaving the vessel. Additionally, the containment would be filled with steam, and fan coolers would create a reflux path. Hence, little vaporization release would be expected, and much of the fission product inventory would be in aqueous solution. Containment failure is not expected during this sequence.

Pressurized-Water Reactor: Small Loss-of-Coolant Accident with Failure of Core Makeup (Failure of Pumped Emergency Core Cooling) (S₂D)

This accident resembles the accident at Three Mile Island 2 except it does not include eventual restoration of core makeup. A small loss-of-coolant accident is initiated by failure of a relief valve, pump seals, or small pipes. System pressure would remain high, unless the accident proceeds to vessel failure, when system pressure would rapidly fall.

At Three Mile Island, the upper regions of the fuel were probably overheated and quenched several times due to the variations in the core makeup flow. This action promoted more-intimate contact of damaged fuel with water and subsequent solution of radionuclides in the water. If the reactor core boils dry with little interruption, the volatile fission product compounds have little opportunity to contact water. However, liquid entrainment during boiling can still create significant contact in the core and primary system.

A factor that must be considered for the class of accidents in which system pressure remains high during and after fuel failure is the effect of high velocities in small piping. Some of these effects are counter-balancing. Enhanced contact with piping walls would encourage chemically active compounds to plate out; however, aerosols or other weakly held substances would be scrubbed off the walls. Also, at the time of melt-through (should the accident progress to that point), the blowdown could be energetic and lightly held materials in the primary system could be dislodged.

Containment sprays and fan coolers would work during this type of accident so any vaporization release would be small, and many of the fission products would be in aqueous solution. Containment failure is not expected in large dry plants but may occur in ice condenser plants if spray cooling is lost.

Pressurized-Water Reactor: Loss of Power or Heat Sinks (TMLB)

The loss of power or heat sinks or both would create high steam pressure that would be relieved through the safety relief valves or the primary overpressure relief valves. Fission product and aerosol behavior in the primary systems would be similar to that of the previously described small loss-of-coolant accident with loss of makeup; both sequences are "wet" with much opportunity for solution of the fission products in water.

On the other hand, the containment building would be relatively dry, since sprays and fan coolers would not work. Radioactive debris could attack concrete and create a substantial vaporization release, particularly after water discharged from an accumulator dump evaporated. The aerosols formed would then tend to deposit and settle on walls, components, and the containment floor. Since containment failure from noncondensable gas and steam overpressure is likely if a heat sink is not reestablished, aerosols could be resuspended during depressurization after containment failure.

Pressurized-Water Reactor: Loss-of-Coolant Accident Caused by Interfacing System Failure (V)

In this loss-of-coolant accident, the check valves that isolate the high-pressure primary system from the low-pressure emergency-core-cooling system fail. The low-pressure system designed for 600 lb/in² pressure is assumed to fail when subjected to primary system pressure of about 2000 lb/in² and present a potential flow path from the core region to the auxiliary building, which is outside containment. Because of the existence of two check valves and their scheduled inspection, this is a low-probability event.

The accident may become serious if core makeup ability is degraded or the water supply is exhausted. Should either occur, the chemical and physical effects on radionuclide transport would be similar to those of an intermediate-break loss-of-coolant accident except the length of pipe the flow must traverse is generally longer and the flow path leads to areas outside containment. This sequence would be wet and would occur under high but gradually decreasing pressure. Velocities in the piping would be high during the blowdown phase before significant fuel failure occurred. After the core boiled down to the point where fuel failure is significant, the velocities would be much lower. For long runs of pipe, plateout would be significant. Because releases in this sequence bypass containment, behavior of radionuclides in the auxiliary building must be investigated.

Boiling-Water Reactor: Large Loss-of-Coolant Accident with Loss of Core Makeup (AE)

Because of the diverse power supply systems and redundant systems for injecting emergency water to the core, this sequence is highly unlikely. Boiling-water reactors have no steam generators. If the steam piping breaks, fast-acting valves isolate the system within containment. If the main steam line breaks, the effluent flow must pass through the steam separators, where significant plateout can occur, as the core boils dry. If a recirculation pipe breaks, the flow would bypass the steam separators and plateout would be less significant. However, almost all boiling-water reactor accidents release core effluents to the suppression pool, where extensive solution of fission product and trapping of aerosols are expected. Since core debris would be released to the drywell upon vessel failure, the debris could attack the concrete and result in a vaporization release.

Boiling-Water Reactor: Small Loss-of-Coolant Accident with Loss of Core Makeup (S₂D)

This accident could be initiated by the failure of a relief valve to close. However, the diverse and redundant systems for supplying core makeup water make the latter part of the event very unlikely. Core depressurization would be rather slow until the automatic depressurization valves opened to reduce the system pressure and allow the high-volume reactor core isolation cooling system to operate. However, it is assumed that this system does not operate or fails once suppression pool temperatures reach a point where the reactor core isolation cooling pumps would fail. It is not clear whether containment would fail before or after the core is uncovered. In the former case, the core could remain covered and no fission products would be released. If core spray fails, however, fuel failure would occur after boildown at relatively low pressure (300 lb/in.²). The fission products would flow past the steam separators, out the valve opening, and through the piping that routes the relief from these valves to the suppression pool. The release of aerosols and soluble radionuclide compounds to the environment would be very small. Fission products released from sparging during concrete attack would be distributed and transported as they are in a large-break loss-of-coolant accident.

Boiling-Water Reactor: Total Loss of Power (TMLB)

This accident sequence assumes that the diverse systems for supplying off-site power and the redundant diesel-drive systems supplying on-site power all fail. Should this sequence be allowed to proceed to completion, the steam-driven reactor core isolation cooling and high-pressure coolant injection pumps would fail because their bearing cooling water comes from the suppression pool, which increases in temperature as a result of failed residual-heat-removal systems. Subsequently, the core would boil dry, releasing fluid through the relief valves and the automatic depressurization valves to the suppression pool. The radionuclide transport would be similar to that of a small loss-of-coolant accident with loss of core makeup, but the suppression pool is hot and less capable of condensing steam. However, it contains large amounts of water available for solution, although both solution and aerosol trapping would be less effective because of mass-transfer rates to the water would be lower. After the reactor vessel failed, a vaporization release from the suppression pool would result from concrete attack in the pedestal and drywell, like similar releases from a small-break or large-break loss-of-coolant accident.

Boiling-Water Reactor: Transient Event with Failure to Scram (TQUV)

In this accident sequence, the reactor protection system fails to scram after main steam isolation valve closure, and standby liquid control is not activated. The recirculation pump trip is successful and the decreased flow through the core increases the average core void fraction, thus decreasing the power.

Given the successful activation of high-pressure injection, the power stabilizes at an average of 20% of rated power due to a balance between the makeup flow and the boil-off rate. The elevated power level causes the suppression pool to heat up rapidly and causes the high-pressure coolant injection and reactor core isolation cooling to fail. The reactor water level boils down and the upper regions of the core are uncovered before the automatic depressurization system, low-pressure coolant injection, and low-pressure core spray are activated. The primary system does not fully depressurize due to the elevated power level, thus reducing the flow capability of low-pressure coolant injection and low-pressure core spray and preventing a reflood. The water level remains at approximately two-thirds of core height until pump suction is lost due to net positive suction head limitations. Core melt and vessel failure follow, resulting in steam overpressurization of containment. Again, a vaporization release would be expected as a result of concrete attack.