UNITED STATES NUCLEAR REGULATORY COMMISSION

POLICY ISSUE
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SECY-97-132

FOR: The Commissioners

FROM: L. Joseph Callan
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SUBJECT: STATUS OF THE INTEGRATION PLAN FOR CLOSURE OF SEVERE ACCIDENT ISSUES AND THE STATUS OF SEVERE ACCIDENT RESEARCH

PURPOSE:

To inform the Commission annually of the status of and progress in implementing the elements of the Integration Plan for the Closure of Severe Accident Issues, i.e., the Individual Plant Examination of Internal Events (IPE), Individual Plant Examination of External Events (IPEEE), Severe Accident Research, and Accident Management (A/M) programs, as requested in a Staff Requirements Memorandum dated April 20, 1989.

SUMMARY:

The Integration Plan for Closure of Severe Accident Issues has four elements and the status is as follows:

IPE Program:

1. All 75 submittals have been reviewed. Staff evaluation reports have been issued for all except five. All submittals except two have been found to be in conformance to Generic Letter 88-20. The Browns Ferry (BF) multi-unit PRA is being reviewed to determine its applicability as an IPE of BF Unit 3.

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2. Draft NUREG-1560 "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance" was published for public comment. An IPE workshop was held in Austin, Texas, on April 7-9, 1997, to solicit and discuss public comments. The final report is to be issued by September 30, 1997.

3. The IPE database was finalized and placed on the NRC web page. Draft NUREG-1603 (IPE database user's manual) was published. A final NUREG-1603 will be issued by December 1997.

4. RES has completed regional briefings on IPE submittals.

5. A program of follow-up activities, based upon IPE insights, is being developed by RES and NRR (September 30, 1997).

IPEEE Program

1. To date, 61 IPEEE submittals have been received with 41 under various stages of review.

2. A preliminary insights report, based on the first 24 IPEEE submittal reviews, is being developed and the report will be available in September 1997.

Severe Accident Research Program

1. The draft report for resolution of the direct containment heating issue for Combustion Engineering and B&W designs has been completed and is undergoing peer review.

2. The results of the Second Steam Explosion Review Group Workshop (SERG-2) were published in NUREG-1524, "A Reassessment of the Potential for an Alpha-Mode Containment Failure and a Review of the Current Understanding of Broader Fuel-Coolant Interaction Issues," in August 1996. The conclusion of the majority of the international experts participating in SERG-2 supported the estimates of low probability of alpha-mode failure (i.e., early containment failure due to internal missiles resulting from an in-vessel steam explosion).

3. Successful experiments as part of the international cooperative RASPLAV and the Melt Attack and Coolability Experiments (MACE) programs have been completed and are providing valuable information in assessing the ability to cool molten core debris either in-vessel or ex-vessel.

4. Lower head failure research also included separate effects testing at Pennsylvania State University and a jointly funded international program to examine gap cooling in the reactor pressure vessel. High pressure creep rupture testing of scaled reactor vessels was also performed.

5. In the area of fission product release and transport, preliminary results from the PHEBUS FPT-1 experiment have provided additional confirmation of the insights reflected in the revised source term.

6. Limited research is continuing in other areas to focus on reducing the uncertainties in select issues in order to improve and maintain the NRC capabilities to analyze severe accident issues.

7. Specific research on hydrogen combustion is underway in support of the staff's review of AP-600 (e.g., testing of a passive autocatalytic recombiner).

8. Code improvement activities are supporting releases of updated versions of the MELCOR and SCDAP/RELAP5 codes and include the completion of the peer review of the VICTORIA code.

9. Detailed analyses of steam generator tube heating during severe accidents were performed to support the ongoing steam generator tube integrity regulatory initiative.

Accident Management Program

1. Licensee implementation of A/M is continuing. Implementation will be completed at approximately 13 sites within the next two months, and an additional 17 sites by late-1997. Implementation at the balance of sites (40) will be completed within the latter half of 1998.

2. The staff attended an industry-sponsored workshop on A/M implementation in March 1997, and expects to participate in a series of licensee "demonstrations" of completed implementation over the next six months. The workshop combined with the A/M demonstrations could serve the role of the information gathering visits described in SECY-96-088, "Status of NRC Assistance to the DOE on Regulatory Plans for Plutonium Disposition Alternatives." The staff intends to reassess and refocus the approach to confirming licensee implementation after the A/M demonstrations.

3. The staff has completed a high level review of the Boiling Water Reactor Owners' Group (BWROG) Emergency Procedure and Severe Accident Guideline documents and identified areas where additional information is needed. The staff anticipates completing review of the BWROG documents by the end of the summer so that BWR licensee schedules for completing A/M implementation will not be impacted.

BACKGROUND:

On May 28, 1988, the staff presented to the Commission the "Integration Plan for Closure of Severe Accident Issues" (SECY-88-147). There were six major elements in that plan: the Severe Accident Research Program (SARP), A/M, Containment Performance Improvement (CPI) and the Improved Plant Operations (IPO) programs. On April 20, 1989, the

Commission requested that the staff provide periodic updates of the status of the various elements of the Plan. The last update was provided in June 1996 (SECY-96-088).

As noted in SECY-95-004, "Status of Implementation Plan for Closure of Severe Accident Issues, Status of Individual Plant Examinations, and Status of Severe Accident Research," and SECY-96-088, the CPI program element has been completed and the Commission is being kept informed of the status of the IPO program through other means. Consequently, the discussion provided below addresses the IPE, IPEEE, Severe Accident Research and A/M programs.

DISCUSSION:

I. IPE Program

1. All 75 IPE submittals have been reviewed. Staff evaluation reports (SERs) have been issued for all except five, two of which are in progress (Susquehanna and St. Lucie) and are expected to be issued by June 30, 1997. The other three submittals have been redone by the licensees to account for either staff concerns brought out during the IPE review process (Byron and Braidwood) or plant changes which resulted in the original IPE submittal being obsolete (Ginna). SERs are scheduled to be issued for these IPEs by the end of July 1997. In addition, all submittals (either the original or a resubmittal) except two (Crystal River and Susquehanna) have been found to be in conformance with the intent of Generic Letter 88-20. Completion of these IPE reviews is expected by the end of December 1997.

2. RES is evaluating the applicability of the TVA Browns Ferry multi-unit PRA, (which is a PRA of Unit 2 given operation of Units 1 and 3) as an IPE of Browns Ferry, Unit 3.

3. Draft NUREG-1560 "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," (Parts 1 and 2) was published in October and November, 1996. Perspectives are presented on four major objectives as follows:

   - The impact on reactor safety;
   - The significant reactor design, containment performance and operational features relative to core damage, containment failure and radionuclide releases;
   - The different methods and models developed and quantified in performing the IPEs; and
   - The implication of the IPE results relative to the Commission's Safety Goals and the Station Blackout Rule.

   A workshop was held in Austin, Texas, on April 7-9, 1996, to present the insights discussed in draft NUREG-1560 and discuss public comments. Approximately 100 participants attended from U.S. power utilities, reactor vendor owners' groups, industry consultants, and other federal and state agencies. Based on comments

received at the workshop and other written comments, a final version of NUREG-1560 will be issued by September 30, 1997.

4. The IPE database has been completed and is available to the public (can be downloaded from the NRC Web page). In addition, draft NUREG-1603, the user's manual for the IPE database, has been published. The final NUREG-1603 will be published by December 1997.

5. RES provided briefings on the IPE results to each of the regions. The briefings were attended by both regional personnel and resident inspectors.

These activities complete the IPE program. However, follow-up activities, based on the insights documented in draft NUREG-1560, are being identified. These activities will be documented in a staff plan as part of the PRA Implementation Plan.

II. IPEEE Program

On June 28, 1991, the NRC issued Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, 10 CFR 50.54(f)," and NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities: Final Report." The generic letter requested all licensees to perform an IPEEE to identify plant-specific vulnerabilities to severe accidents caused by external events and report the results to the NRC.

To date, the staff has received 61 IPEEE submittals and will receive an additional 11 by the end of 1997, one by June 1998, and one with a date not yet determined. Currently, 41 submittals are under various stages of review with these reviews performed primarily with contractor support and reviewed by a senior review board of staff and contractors expert in PRA, fire and seismic analyses, as well as other relevant disciplines. The IPEEE review process focuses on: (a) quality and completeness of the submittals and (b) assessments and resolution of certain generic issues (see Attachment 1). An SER will be issued following the completion of each review indicating whether or not the submittal has met the intent of Generic Letter 88-20 and adequately addressed the relevant generic issues.

As stated in SECY-96-088, the staff is preparing a preliminary insights report based on the first 24 IPEEE submittal reviews. This report will: (1) summarize the significant IPEEE findings and evaluate whether any generic observations can be derived, (2) evaluate lessons learned about the methodologies used, and (3) assess the usefulness of the IPEEE analyses for regulatory applications. This report will be available by the end of September 1997. Some of the preliminary insights, together with newly received information from the Quad Cities IPEEE submittal, are:

1. Core damage frequencies (CDFs) due to internal fires range from ~2E-9 to 5E-3 per reactor year (RY), (fire CDF for Quad Cities is reported to be ~5E-3/RY);

2. CDFs due to seismic events range from ~1E-7 to 2E-4/RY and seismic capacities, i.e., in terms of high confidence, low probability of failure, range from 0.05 to 0.5g (peak ground acceleration);

3. CDFs due to external floods and high winds range from less than 1E-8 to 6E-5/RY;

4. The fire CDF (about 5E-3/RY) at Quad Cities and the seismic CDF (about 2E-4/RY) at Haddam Neck were considered by the licensees to be potential vulnerabilities. Commonwealth Edison, the licensee for Quad Cities, has implemented some interim measures to reduce the potential fire risk and currently is evaluating additional measures to further reduce the potential fire risk. Northeast Utilities (NU), the licensee for Haddam Neck, had made many significant plant-specific improvements before submitting its IPEEE and stated in its submittal that it planned to assess what additional improvements would be needed to further reduce the potential seismic risk. (Since then, NU has decided to permanently shut down this plant.)

5. At many other plants, the licensees did not report potential vulnerabilities associated with external events, however, many plant-specific improvements were implemented at those plants.

III. Severe Accident Research Program

The Severe Accident Research Program has provided support for the certification review of AP600 and has focused on phenomena and issues to understand and quantify potential challenges to containment integrity, with particular emphasis placed on addressing early containment challenges. Significant progress has been made in resolving both the direct containment heating and alpha-mode failure issues. Research is also underway to provide a better understanding of issues regarding molten core debris coolability, which ultimately may provide accident management strategies which can mitigate potential containment challenges. A limited number of experimental programs conducted at universities or under cooperative international agreements will continue to focus on specific issues, such as hydrogen combustion and fission product release, chemistry, and transport, and will support maintenance of expertise. The results of the experimental programs are used to develop and validate improved models in the NRC's severe accident codes. In many areas, the NRC has participated and will continue to participate in jointly funded cooperative projects with industry and foreign countries and organizations in order to leverage NRC resources. The status of the specific research areas is discussed in Attachment 2.

IV. Accident Management Program

The goal of the accident management (A/M) program is to enhance the capabilities of the licensee's Emergency Response Organization (ERO) to prevent and mitigate severe accidents and minimize any off-site releases. As part of A/M implementation, the insights developed through the conduct of the IPEs, such as important accident sequences and equipment/system failure modes, will be considered by licensees in their development and implementation of plant-specific severe accident management guidance and ERO personnel training program enhancements.

In SECY-96-088, the staff described the industry commitment and schedules for implementing A/M pursuant to a formal industry position on this matter, staff plans for confirming the adequacy of licensee implementation, and the status of the review of severe accident management guidance for BWRs. Significant progress has been made since SECY-96-088 was issued, as described below.

Licensee Implementation of A/M

As described in SECY-96-088, all licensees have committed to implement A/M in accordance with the formal industry position documented in Revision 1 to Nuclear Energy Institute (NEI) 91-04, "Severe Accident Issue Closure Guidelines," and have provided target dates for completing implementation. Although several licensees have reported some schedule slippage in the interim period, the schedules for completion are largely unchanged from the original commitment dates.

Licensee implementation of A/M is proceeding. Implementation will be completed at approximately 13 sites within the next two months and an additional 17 sites by late-1997. Implementation at the balance of sites (40), including the majority of the BWR sites, will be completed within the latter half of 1998. The later completion dates for BWRs are due to a BWROG decision to integrate the severe accident guidance within the Emergency Operating Procedures and delays in completing development of the integrated Emergency Procedure and Severe Accident Guideline package.

Plans for Evaluating Licensee Implementation

In SECY-96-088, the staff outlined plans to perform a limited number of pilot inspections to develop confidence in licensee A/M implementation, combined with less detailed evaluations of A/M performance for the balance of plants. Major steps in the staff's approach for evaluating licensee implementation included: (1) conducting information gathering visits at two to four sites to observe how the elements of the formal industry position are being implemented, (2) completing a temporary instruction (TI) using insights obtained through the site visits, (3) performing pilot inspections at about five plants using the TI, (4) developing an inspection procedure (IP) for use at remaining plants based on findings from the pilot inspections and feedback from industry, (5) evaluating implementation at remaining plants using the IP, and (6) in the longer term, evaluating A/M maintenance on a for-cause basis as a regional initiative.

The staff met with NEI on December 19, 1996 to discuss the scope and schedules of the information gathering visits. At that time, NEI proposed to take the lead in organizing "demonstrations" of completed A/M implementation at four to six plants. These demonstrations would be in lieu of the information gathering visits and follow-on pilot inspections envisioned by the staff and would occur in the mid-1997 time frame. NEI also informed the staff of an industry-sponsored workshop concerning severe accident management implementation planned for March 11-13, 1997, and proposed that NRC staff attend in order to better understand the implementation approach and status.

In a follow-up meeting with NEI on January 24, 1997, the staff indicated that attendance at the A/M workshop together with participation in the A/M demonstrations could serve

the role of the information gathering visits, but that changes to the plans outlined in SECY-96-088 concerning the need for pilot inspections and the nature of the inspections at the balance of plants are not warranted at this time. The staff intends to reassess and refocus this aspect of the program after the A/M demonstrations.

NRR staff attended the NEI-sponsored workshop on accident management implementation on March 11-13, 1997. The purpose of the workshop was to provide a forum for utility personnel to explore and discuss alternative solutions to issues that have arisen during plant-specific implementation of severe accident management guidance (SAMG) and training. The workshop was open to the public and attended by approximately 200 persons, including utility staff responsible for implementing the various facets of severe accident management at their plants, representatives from each of the owners groups, the Electric Power Research Institute (EPRI), and several foreign organizations. The workshop provided the staff an opportunity to better understand plant-specific implementation approaches and issues and the major elements of implementation, such as development of plant-specific SAMG, initial staff training, SAMG validation, conduct of A/M drills and tabletop exercises, and use/applicability of 10 CFR Part 50.59 in the implementation process.

The staff is currently awaiting confirmation from NEI regarding the schedule and locations of the plant-specific A/M demonstrations. The first A/M demonstration visit is tentatively planned for late May 1997. A second demonstration visit is also being considered for late July 1997.

BWR Emergency Procedure and Severe Accident Guidelines

In SECY-96-088, the staff described the submittal of severe accident management guidance documents by the Boiling Water Reactor Owners Group (BWROG) and the initiation of a high-level review of the BWR Emergency Procedure and Severe Accident Guidelines (EP/SAG). (Severe accident management guideline documents have already been submitted by each of the PWR owners groups and reviewed by the staff, as described in SECY-94-166, "Status of Implementation Plan for Closure of Severe Accident Issues, Status of Individual Plant Examinations, and Status of Severe Accident Research.")

Subsequently, the BWROG submitted Rev. 0 of the (EP/SAG) and associated technical basis documents to NRC for information on August 29, 1996. The staff and Oak Ridge National Laboratory have completed a high level review of the EP/SAG documents. Areas where additional information and discussion with the BWROG is considered necessary were identified in an April 2, 1997, letter to the owners group. The BWROG has agreed to illustrate the EP/SAG implementation process and time-line by applying the guidelines to a limited number of BWR sequences identified by the NRC. A submittal from the BWROG containing this information is expected shortly. A meeting to discuss specific questions/concerns regarding the BWROG products will be scheduled once the submittal is received and the BWROG is prepared to address staff concerns. The staff anticipates completing review of the BWROG documents by the end of the summer so that BWR licensee schedules for completing A/M implementation will not be impacted.

The staff will continue to keep the Commission informed of progress on the above areas, An update of this paper on the status of the Integration Plan for the Closure of Severe Accident Issues will be provided in May 1998.

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Attachments:
1. Resolutions of Generic Safety Issues Dependent on IPEEE Reviews
2. Severe Accident Research Program

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RESOLUTIONS OF GENERIC SAFETY ISSUES DEPENDENT ON IPEEE REVIEWS

Resolutions of the following generic safety issues are dependent on the reviews of IPEEE submittals. These generic issues are of the following two types:

(1) Issues identified during the initial planning of the IPEEE program and explicitly discussed in Supplement 4 of GL 88-20. These include:

- Unresolved Safety Issue A-45: Shutdown decay heat removal requirements
- Generic Issue 131: Potential seismic interaction involving the movable in-core flux mapping system used in Westinghouse plants (portions of the system have not been seismically analyzed)
- Eastern U.S. Seismicity: Charleston earthquake issue - eight plants identified as needing additional review
- Fire Risk Scoping Study: Plant-specific analyses needed to assess risk importance of certain fire risk issues
- Generic Issue 103: Design for probable maximum precipitation

(2) Issues addressed by the staff subsequent to the issuance of Supplement 4 of GL 88-20, the resolution of which was connected to the plant-specific analyses being performed in the IPEEE program. These include:

- Generic Issue 156, "Systematic Evaluation Program (SEP)": Nine issues related to seismic-, fire-, and flood-initiated accidents were identified as being resolved as part of IPEEE
- Generic Issues 147 and 148: "Fire-Induced Alternate Shutdown/Control Room Panel Interactions," and "Smoke Control and Manual Fire-Fighting Effectiveness," respectively
- Generic Issue 57, "Effects of Fire Protection System Actuation on Safety-related Equipment"
- Generic Issue 172, "Multiple System Response Program": Eleven issues related to seismic-, fire-, and flood-initiated accidents were identified as being resolved because they will be addressed in IPEEE.

The issues identified above, except the Eastern U.S. Seismicity and SEP issues, apply to all plants. The Eastern U.S. Seismicity and SEP issues apply only to certain plants submitting an IPEEE. Table 1 identifies the plants for which these specific issues apply.

Table 1 IPEEE Submittals
Requiring Review of EUS SEISMICITY and SEP Issues

Plant	SEP Plant	Eastern U.S. Seismic Plant
D. C. COOK 1-2	X	
KEWAUNEE	X	
HADDAM NECK	X	
TURKEY POINT 1-2	X	
BIG ROCK POINT	X	
BRUNSWICK 1-2	X	
FORT CALHOUN	X	
PALISADES	X	
PILGRIM	X	X
POINT BEACH 1-2	X	
ROBINSON	X	
THREE MILE ISLAND	X	
INDIAN POINT 2	X	X
DUANE ARNOLD	X	
MONTICELLO	X	
PEACH BOTTOM 1-2	X	
HATCH 1-2	X	
MAINE YANKEE	X	
MILLSTONE 2	X	
OYSTER CREEK	X	
VERMONT YANKEE	X	
OCONEE 1-3	X	X
ARKANSAS 1-2	X	X
FITZPATRICK	X	
CALVERT CLIFF 1-2	X	

Plant	SEP Plant	Eastern U.S. Seismic Plant
MILLSTONE 1	X	
BROWNS FERRY 1-3	X	
GINNA	X	
COOPER	X	
NINE MILE POINT 1	X	
SURRY 1-2	X	
PRAIRIE ISLAND 1-2	X	
QUAD CITIES 1-2	X	
ZION 1-2	X	
INDIAN POINT 3	X	X
DRESDEN 2-3	X	

The status of the specific research areas is discussed below:

Direct Containment Heating: Direct Containment Heating (DCH) refers to the process whereby, under certain accident scenarios, molten core debris is ejected under high pressure from the reactor vessel into the containment atmosphere. The subsequent rapid heating of the containment atmosphere, in conjunction with possible hydrogen combustion, can lead to early containment failure. DCH was identified as one of the important contributors to early containment failure for PWRs in NUREG-1150 and has also been identified as one of the leading contributors to early containment failure for PWRs in the IPEs. The results of previous research into the characteristics of debris dispersal and resultant containment loadings has led to closure of the DCH issue for all Westinghouse plants with large dry or subatmospheric containments, excluding ice condenser plants. Using a probabilistic framework to address uncertainties in the estimate of containment loads, the analysis leads to the conclusion that the containment is not threatened by credible loads resulting from a high pressure melt ejection (NUREG/CR-6338, February 1996). A draft analysis to address DCH in ice condenser plants is nearing completion and will undergo a peer review. The peer review process is expected to be completed by the end of 1997.

The DCH issue resolution methodology, which was previously used for Westinghouse plants, is also being used to address the DCH issue for the large dry reactor containments of the Combustion Engineering and Babcock and Wilcox designs. The model used to calculate containment loads due to DCH for these plant designs was bench-marked against large scale integral tests, conducted at Sandia National Laboratories, to investigate DCH in a CE-like design similar to that of Calvert Cliffs. (The CE-like designs have a reactor cavity design that results in a greater dispersal of the core debris into the containment atmosphere than do the cavities of the Westinghouse designs.) The results of these large scale DCH integral tests for CE-like designs were published in NUREG/CR-6469, "Experiments to Investigate Direct Containment Heating Phenomena with Scaled Models of the Calvert Cliffs Nuclear Power Plant," in February 1997. Preliminary results of the DCH issue resolution for CE and B&W plants are documented in draft NUREG/CR-6475. Peer review of NUREG/CR-6475 is near completion, and the final report will be published by September 1997.

Fuel-Coolant Interactions and Debris Coolability: NUREG-1150 and some IPEs have identified energetic fuel-coolant interactions (FCIs) or steam explosions as important contributors to early containment failure. In NUREG-1150, the alpha mode failure of the containment resulting from in-vessel steam explosions represented a significant fraction of the early failure probability for the Surry and Zion plants (although the overall likelihood of early containment failure was low). In June 1995, the Second Steam Explosion Review Group Workshop (SERG-2) was held to review the status of FCI research. The results of this review meeting were published in NUREG-1524, "A Reassessment of the Potential for an Alpha-Mode Containment Failure and a Review of the Current Understanding of Broader Fuel-Coolant Interaction Issues," in August 1996. The overall conclusion of the majority of the international experts participating in SERG-2 was that alpha-mode failure was a very low probability event and therefore resolved from a "risk perspective."

While the issue of alpha-mode failure is considered resolved as determined by the SERG-2, the experts did recommend a number of areas of future research relating to broader FCI issues. These areas would improve the ability to analyze other FCI challenges such as steam explosions in the lower reactor head or in ex-vessel reactor cavities. In this regard, under the Technical Exchange Arrangement between the NRC and Commission of the European Communities Joint Research Center (JRC), in Ispra, Italy, experimental work is being conducted as part of the FARO and KROTOS programs, using prototypic materials, to address the range of fuel-coolant interactions associated with both in-vessel and ex-vessel accident progression. An experimental program is also underway at Argonne National Laboratory to explore the chemical augmentation of fuel-coolant interactions using reactor materials (Zr and ZrO_2). Finally, small scale experiments at the University of Wisconsin, using stimulant materials, are examining issues involving the energetics of steam explosions.

In the area of debris coolability, RES is participating in a cooperative program, with EPRI, Department of Energy (DOE), and a number of international regulatory and research organizations, called the Melt Attack and Coolability Experiments (MACE) program. This program's objective is to determine the ability of water to cool prototypic ex-vessel core debris, thereby preventing basemat meltthrough. In January 1997, the M3b test, using 2000 kg of prototypic material, was successfully completed. Preliminary results have identified several cooling mechanisms that were involved in cooling the molten debris. Further analysis is underway. Discussions are planned with program participants regarding continuation of the program.

Hydrogen Combustion: In support of NRR's review of the AP600, two series of tests were completed at Sandia National Laboratories (SNL) to evaluate the performance of passive autocatalytic recombiners (PARs). Westinghouse proposes to use PARs in the AP600 design for the control of combustible gases following a design basis accident. Preliminary test results at SNL confirm the PARs' ability to recombine hydrogen with oxygen at relatively low concentrations (below 1% mole hydrogen) in both hydrogen-air and hydrogen-air-steam environments. Additional tests are planned to further explore the ignition potential of PARs and to better characterize the performance of PARs.

RES is also participating in two international cooperative programs aimed at extending the data base on hydrogen combustion into more prototypic situations. Under a cooperative program with NUPEC of Japan, testing was performed for detonation transmission in the large scale high temperature combustion facility at Brookhaven National Laboratory (BNL). This work is aimed at establishing criteria for detonation transmission of hydrogen-air-steam mixtures at elevated temperatures (500-700K). In another program, the NRC, FZK of Germany, and IPSN of France are coordinating an experimental program at the Russian Research Center (RRC) to investigate hydrogen combustion issues at large scale. These large scale experiments are being performed to study deflagration to detonation transition (DDT) in a steam environment and to verify hydrogen igniter separation distance. The findings from these experiments are being used to develop a generalized methodology to predict the possibility of detonations due to DDT in hydrogen, air, steam mixtures. Finally, small scale experiments are continuing at the California Institute of Technology to study diffusion flame stability and expansion of high speed jets into hydrogen mixtures.

Lower Head Failure/Vessel Integrity: One area of research of considerable interest worldwide over the last several years is to determine whether, during a severe accident,

molten core debris can be retained in-vessel, through either in-vessel cooling or ex-vessel cooling by flooding the reactor cavity. The NRC is cooperating with 14 countries under the auspices of the Organization for Economic Cooperation and Development's (OECD) Nuclear Energy Agency (NEA) to investigate melt-vessel interactions to provide data on the internal natural convection flow and local heat flux distribution inside the lower head of the reactor pressure vessel (RPV) for various melt compositions. This program involves large-scale integral experiments using molten UO_2, Zr, and ZrO_2 (corium) in representative reactor lower head geometries, analytical studies, and a number of small-scale separate effects experiments. This program, named RASPLAV, is being performed at the Russian Research Center, Kurchatov Institute. In October 1996, the first successful large scale experiment with 200 kg of corium was performed. During this test the corium temperature reached 2700°C, and natural convection in the corium was established. Extensive post-test examination of the ingot from this experiment is currently underway. In May 1997 the second large scale test was performed. Although the test was terminated early (one hour into a planned 4-hour test), preliminary indications are that sufficient data was obtained to provide useful results.

A small scale experimental program is under way at Pennsylvania State University to address ex-vessel flooding of the reactor cavity to prevent vessel failure. The program investigates boiling heat transfer on downward facing surfaces in hemispherical and toroidal geometries. The results of these experiments have provided data on the critical heat flux (CHF) distribution on the bottom curved surface of the reactor vessel which led to the development of an analytical model for CHF on downward facing surfaces. Experiments on the effect of insulation, similar to that proposed for the AP600 design, are presently being performed and will be completed later this year.

Research is also underway to examine the possibility of cooling molten core debris through in-vessel cooling. In July 1996, Phase I of a cooperative experimental program on in-vessel debris coolability was completed at Fauske and Associates, Inc. (FAI). This project is jointly funded by the NRC, EPRI and organizations in Japan, France and Sweden. Four scaled experiments were completed during Phase I using simulant material (Fe/Al_2O_3). The results of these experiments demonstrated that, with water present, molten material does not adhere to the vessel wall, and the vessel wall can strain away from the debris crust, thereby creating a gap that can enhance cooling of the debris and the vessel wall. Phase II of this program is currently being conducted using an oxidic simulant debris (Al_2O_3) under various pressure and initial conditions. Also in support of the concept of in-vessel cooling, an experimental test facility was designed and built at the Russian Research Center to investigate heat transfer in gaps formed between the corium crust and the inner boundary of the lower head of the RPV. In this experimental test setup, the CHF is measured in gaps which are formed by two vertical walls. Test with non-vertical walls are planned.

Finally, an experimental program is ongoing at Sandia National Laboratories to better understand the mode, mechanism, location, timing, and characteristics of the failure of a reactor pressure vessel lower head under the combined effects of thermal and pressure loads if the molten core debris can not be cooled in-vessel. The first four experiments in this program were completed using scaled lower head test sections by October 1996. These experiments investigated lower head failure with both local and global heating and with and without vessel penetrations. Preparations for additional experiments to examine failure at lower system pressure are currently underway. The results of these experiments will be used to develop improved models of RPV failure in NRC's severe accident codes.

Fission Product Release, Chemistry and Transport: Research in this area is primarily through the participation in the PHEBUS-FP (fission product) project. The PHEBUS-FP project, sponsored jointly by the Commissariat a l' Energie Atomique and the Commission of the European Communities with participation by the NRC under a cooperative agreement, is aimed at studying accident progression and fission product behavior in the reactor system and containment. On July 26, 1996, the second integral Phebus test, Phebus FPT-1, was conducted. This test, similar to FPT-0, was different in that it was conducted with pre-irradiated fuel with a correspondingly much larger fission product inventory. The test involved the melting of approximately 30% of the fuel and the release of over 70% of the volatile fission products. Preliminary data indicate that approximately 25% of the initial core inventories of iodine and cesium were transported to the containment. Only trace amounts of iodine were detected as gaseous iodine in the containment, confirming the insights reflected in the NRC's revised source term as described in NUREG-1465. Additionally, iodine in the sump was detected as an insoluble species, Ag I, and it was concluded that little or no revolitization of iodine by radiolysis took place. The results of the FPT-1 test, and its predecessor FPT-0, have been extensively used for the assessment and validation of NRC severe accident codes. The next test, FPT-4, will examine fission product releases from a fuel debris bed rather than an initially intact fuel geometry. This will provide insights on the releases from accidents where the fuel is fragmented prior to significant melting.

Code Development/Improvement: Because of the difficulty in performing prototypic experiments for a variety of severe accident scenarios, substantial reliance must be placed on the development, verification, and validation of computer codes for analyzing severe accident phenomena. The severe accident codes provide the staff the analytical tools necessary to model plant accidents and transients to assist in resolving safety issues and for incorporating research results into the regulatory process. In the area of severe accident code development and assessment, a number of important activities should be noted.

MELCOR, the full-plant systems-level severe accident code, has been significantly enhanced, and an updated version, MELCOR 1.8.4, will be released in June 1997. Currently, MELCOR is used in conjunction with international cooperative experiments such as PHEBUS and an array of plant analyses associated with specific risk evaluations. Further, MELCOR is one of the most widely-used severe accident codes in the world. As such, RES supports the MELCOR Cooperative Assessment Program, an international program to promote the exchange of MELCOR assessment information and to provide the NRC with feedback concerning the use of the code by others.

Significant progress has also been made to implement improved models into the SCDAP/RELAP5 code. This code is a detailed mechanistic code for analysis of in-vessel severe accident progression for conventional plants (both PWRs and BWRs) and advanced light water reactor plants (ALWRs) from the initial phases of an accident, through core uncovery, core degradation and relocation, and to reactor vessel or system failure. The MOD3.2 version of SCDAP/RELAP5 is scheduled for release by the end of September 1997. The SCDAP/RELAP5 code has been used to support the review of in-vessel coolability and retention of a core melt for the AP600 design.

The VICTORIA code, a mechanistic fission-product-behavior code for analyzing fission-product release and transport in the reactor coolant system, has recently undergone an

independent peer review. The peer review committee, which consisted of fission-product behavior experts from the U.S. and France, identified specific recommendations for code improvement. The committee's findings are currently being addressed as part of the plan for code maintenance. The peer review of the fuel-coolant interaction code, IFCI, was also completed. Finally, RES has initiated a cooperative project with FZK in Germany and DOE (through Los Alamos National Laboratory) to support the development and assessment of the GASFLOW code, a multi-dimensional (3-D) finite volume field code capable of predicting post-accident local conditions inside containment.

Steam Generator Tube Integrity: During the past year, a significant effort was devoted to the support of the proposed rulemaking on steam generator tube integrity. This included a comprehensive examination of the thermal hydraulic boundary conditions imposed on steam generator tubes during a limiting severe accident scenario. The scenario analyzed was a high pressure sequence (station blackout) further aggravated by the assumption that the steam generator is depressurized by failure, in the open position, of secondary system relief valves. Analyses were performed using the suite of severe accident codes, SCDAP/RELAP5, VICTORIA and MELCOR. The bulk of the effort went into examination of high temperature vapor circulation through the steam generator tubes using the SCDAP/RELAP5 code. VICTORIA analyses were performed to examine the effects of fission product deposition on the tubes and the heating of tubes by this mechanism. MELCOR analyses were performed to assess the offsite dose consequences for assumed tube leakage rates and tube ruptures. These assessments were then factored into the overall risk assessment considering the impact of steam generator tube defects.