June 8, 1998 SECY-98-131

FOR: The Commissioners

FROM: L. Joseph Callan /s/
Executive Director for Operations

SUBJECT: STATUS OF THE INTEGRATION PLAN FOR CLOSURE OF SEVERE ACCIDENT ISSUES AND THE STATUS OF SEVERE ACCIDENT RESEARCH

PURPOSE:

To inform the Commission of the status 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, and to recommend an alternate approach to keep the Commission informed of progress in these areas.

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: IPE, IPEEE, Severe Accident Research, A/M, Containment Performance Improvement and the Improved Plant Operations 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 on June 23, 1997 (SECY-97-132).

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As noted in SECY-97-132, the Containment Performance Improvement program element has been completed and the Commission is being kept informed of the status of the Improved Plant Operations program through other means. Consequently, the discussion provided below addresses the IPE, IPEEE, Severe Accident Research, and A/M programs.

DISCUSSION:

I. IPE Program

On November 23, 1988, NRC issued Generic Letter 88-20, “Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR 50.54 (f),” and NUREG-1335, “Individual Plant Examination Submittal Guidance.” The generic letter requested all licensees to perform a systematic examination to identify plant-specific vulnerabilities to severe accidents and report the results to the Commission.

1. The staff has reviewed all of the 76 IPE submittals and issued staff evaluation reports (SERs) on its findings to each licensee with one exception, Browns Ferry 3 (BF3). In three of the last SERs, the staff has not been able to conclude that the licensee met the intent of Generic Letter 88-20 for their plant(s). These three IPEs include Crystal River 3, Susquehanna 1&2, and BF3. Recently, the staff met with the licensees for both Crystal River 3 and Susquehanna 1&2. It appears that these licensees have addressed the staff concerns, and the staff will issue updated SERs by the end of June 1998. RES forwarded to NRR a draft SER on BF3 in December 1997, which will be forwarded to the licensee in the near future. NRR will then arrange a meeting with the licensee to discuss resolution of the BF3 IPE.

2. The final version of NUREG-1560, “IPE Program: Perspectives on Reactor Safety Plant Performance,” was issued in October 1997. IPE follow-up activities are included in the PRA Implementation Plan.

II. IPEEE Program

On June 28, 1991, 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 70 IPEEE submittals. Two additional submittals are expected by June 1998, and the date for two submittals has yet to be determined. Currently, 65 submittals are under various stages of review. These reviews are initially performed by NRC staff and contractors and are subsequently 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. An SER will be issued following the completion of each review indicating whether or not the submittal has met the intent of Supplement 4 to Generic Letter 88-20 and adequately addressed the relevant generic issues. One SER on Diablo Canyon was issued to the licensee in December 1997.

The staff has prepared a preliminary insights report based on the first 24 IPEEE submittal reviews. This report discusses the significant IPEEE findings and generic observations as well as lessons learned about the methodologies used. This report was sent to the Commission on January 20, 1998.

III. Severe Accident Research Program

As many of the experimental programs have been either completed or are being terminated due to budget constraints, the Severe Accident Research Program is focusing its activities on the maintenance of analytical capabilities (i.e., severe accident codes) necessary to support risk informed regulatory initiatives, support resolution of severe accident issues, and support accident management decisions. This includes increasing the in-house analytical capabilities to perform analysis previously performed at contractor organizations. A limited number of experimental and analytical activities, however, are continuing to address those areas of risk significance with the largest uncertainty, with particular emphasis placed on addressing early containment challenges. In this regard, progress has been made in the resolution of the direct containment heating issue and research is 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. These remaining experimental programs are jointly funded cooperative projects with industry and foreign countries, which allow the NRC to leverage its resources. Finally, the Severe Accident Research Program has provided support for the certification review of AP600 and is currently supporting the implementation of the revised source term. The status of the major areas of research is discussed in Attachment 1.

IV. Accident Management

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, licensees will consider the insights developed through the conduct of the IPEs, such as important accident sequences and equipment/system failure modes, when they develop and implement plant-specific Severe Accident Management Guidelines (SAMG) and ERO personnel training program enhancements.

In SECY-97-132, the staff described the status and schedule for licensee implementation of A/M plans for evaluating licensee implementation, and the status of the staff’s review of the BWROG Emergency Procedure and Severe Accident Guidelines. Significant progress has been made since then, as licensee implementation of A/M is progressing. Implementation will be completed at approximately 20 sites by summer 1998, and at the remaining sites by the end of 1998. The A/M demonstration phase was completed in March 1998, and included visits at four sites. The demonstrations provided insights into the licensee's A/M implementation and evaluation process. However, the staff identified several areas important to effective A/M implementation where additional staff evaluation is needed. Accordingly, the staff intends to proceed with the pilot inspections described in SECY-96-088 and SECY-97-132. NRC and NEI are continuing to communicate on a number of implementation issues identified by licensees. Finally, the staff has taken several steps to disseminate information regarding A/M program implementation and products, including involving staff from the regional offices in the A/M demonstration visits, and providing oversight training on the Westinghouse Owners Group SAMG for members of the headquarters Reactor Safety Team. A more detailed status is discussed in Attachment 2.

RECOMMENDATION:

The staff currently reports to the Commission on the progress of the IPE, IPEEE, and A/M as part of the Quarterly Status Report on the Probabilistic Risk Assessment Implementation Plan. The most recent report (SECY-98-096) was provided to the Commission on May 1, 1998. Because the Commission is already provided progress of these activities through the Quarterly Status Report and as a result of the completion of a number of the severe accident research program activities, the need for this annual update has diminished. Therefore, unless instructed otherwise, this will be the last annual update. When significant issues arise from the remaining severe accident research activities, the staff will inform the Commission at that time.

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Attachment: 1. Severe Accident Research Program Status
2. Accident Management Status

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Attachment 1

Severe Accident Research Program Status

Status of major areas of research is discussed below:

Hydrogen Combustion: In support of NRR's review of the AP600, an experimental program was 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. The SNL program was used to perform confirmatory tests to assess PAR performance under a range of hydrogen, air, and steam conditions. Results from these tests 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. However, these tests also indicate the potential for PARs to ignite mixtures at hydrogen concentrations in the range of 5-7%.

During the past year, RES has also participated 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 has been performed for detonation transmission at elevated temperatures (500K-700K) at the high temperature combustion facility at Brookhaven National Laboratory (BNL). The remaining work, aimed at establishing criteria for detonation transmission of hydrogen-air-steam mixtures at elevated temperatures (500-700K), will be completed by September 1998, at which time this program will be closed. In another program, the NRC, Forschungszentrum Karlsruhe (FzK) in Germany, and Institute de Protection et de Surete Nucleire of the Commissariat a l'Energie Atomique (IPSN) in France coordinated an experimental program at the RUT facility at the Russian Research Center (RRC) to investigate hydrogen combustion issues at large scale. These large scale experiments provided data to develop a generalized methodology to predict the possibility of detonations due to deflagration-to-detonation transitions in hydrogen, air, steam mixtures. With the completion of these experiments in December 1997, NRC's participation in this cooperative program ended.

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 previous analysis led to the conclusion that the containment is not threatened by credible loads resulting from a high pressure melt ejection (NUREG/CR-6338, February 1996). Using the same methodology, a draft analyses to address DCH in ice condenser plants has been completed and recently underwent a peer review. Based on this draft analysis, it has been found that ice condenser plants do not have the same inherent capacity to withstand the credible DCH loads from all scenarios as other

Westinghouse plants. As a result, and as recommended by the peer review, additional analyses are being performed to better characterize the likelihood of high pressure melt ejection (HPME) scenarios for ice condenser plants and to refine the load/strength analyses for these containment designs, with the objective to determine whether the likelihood of DCH failing the containment is sufficiently low that this issue can be considered resolved for these plants. Completion of this effort is expected by the end of 1998.

The DCH issue resolution methodology, which was previously used for Westinghouse plants, has also been used to address the DCH issue for the large dry reactor containments of the Combustion Engineering and Babcock and Wilcox designs. Results of the DCH issue resolution for CE and B&W plants, which addresses 15 CE plants and 7 B&W plants, indicate that the conditional containment failure probability for accident scenarios that can lead to HPME is less than 0.1 and that DCH can be considered resolved for all CE and B&W plants. Peer review of the NUREG/CR report has been completed and the report is expected to be published by June 1998. Finally, two supplemental DCH tests are being performed at SNL under a cooperative program with FzK and IPSN. The tests are being performed at lower reactor coolant system pressures than previous tests to provide data regarding intentional depressurization as an accident management strategy to mitigate DCH loads. These tests are expected to be complete by June 1998.

<u>Fuel-Coolant Interactions:</u> Both NUREG-1150 and more recent IPEs have identified energetic fuel-coolant interactions (FCI) or steam explosions as important contributors to early containment failure. To obtain data necessary for the assessment of analytical tools used for FCI analyses, the NRC is continuing its participation in the FARO and KROTOS programs being conducted at Commission of the European Communities Joint Research Center (JRC) in Ispra, Italy. The FARO program is providing data for determining quenching and long-term coolability of core debris. The KROTOS program is providing data for determining the conditions under which steam explosions can occur. Both programs are using prototypic materials in the experimental program. The remaining experimental programs at Argonne National Laboratory, to explore the chemical augmentation of fuel-coolant interactions using reactor materials (Zr and ZrO_2), and the small scale experiment at the University of Wisconsin, using simulant materials to examining issues involving the energetics of steam explosions, are being terminated.

<u>Lower Head Failure/Vessel Integrity:</u> One area of research of continuing interest worldwide is to determine whether, during a severe accident, molten core debris can be retained in-vessel, either through in-vessel cooling or ex-vessel cooling by flooding the reactor cavity. In this regard, the NRC is continuing its cooperation with 17 countries under the auspices of the Organization for Economic Cooperation and Development's (OECD) Nuclear Energy Agency to investigate melt-vessel interactions, molten corium pool chemical behavior and to provide data on the internal natural convection flow and local heat flux distribution inside the lower head of the reactor pressure vessel for various prototypic corium melt compositions. This program involves large-scale integral experiments using molten UO_2 and ZrO_2 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. Three successful large scale experiments, conducted with 200 kg of corium, have been performed. During these test the corium temperature has reached as high as 2700°C, and natural convection in the corium has been established. In the first two tests, stratification of

melt into two layers was observed with a Zr enriched layer on the top. The third test was completed on April 7, 1998, and post-test examination of this experiment is currently underway.

The current experimental program 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 is nearing completion. Eight experiments were completed on a scaled lower head test section, with the last experiment completed in March 1998. These experiments investigated lower head failure with various heat flux distributions, with and without vessel penetrations. The results of these experiments are being used to develop improved models of RPV failure. Because of the international interest in this program, eight additional experiments are currently being proposed as an OECD sponsored project.

In a project jointly funded by the NRC, EPRI and organizations in Japan, France and Sweden, research is also underway to examine the possibility of cooling molten core debris through in-vessel cooling. Phase II of this program is currently being conducted at Fauske and Associates, Inc., using an oxidic simulant debris (Al_2O_3) under various pressure and initial conditions, to explore whether, with water present, molten material will not adhere to the vessel wall, and the vessel wall will strain away from the debris crust, thereby creating a gap that can enhance cooling of the debris and the vessel wall. This phase of the program, and NRC participation, will be complete by August 1998. Finally, the small scale experiments, conducted at Pennsylvania State University, to address ex-vessel flooding of the reactor cavity to prevent vessel failure have also been completed. This program provided data on the critical heat flux (CHF) distribution on the bottom curved surface of the reactor vessel and the effect of insulation, similar to that proposed for the AP600 design, and has provided information to support the certification review of the AP600.

<u>Fission Product Release, Chemistry and Transport:</u> Research in this area is 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 the 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. The test involved the melting of approximately 30% of the fuel and the release of over 70% of the volatile fission products. Results to date 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. Additionally, iodine in the sump was detected as an insoluble species, Ag I, thus 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.

Staff expertise developed in this area is currently being used to support the implementation of the revised source term. In October 1997, RES took over the lead responsibility for completing an integrated assessment of the impact of the use of the revised for operating plants, referred to as “rebaselining.” The staff will be providing the Commission the results of this rebaselining in a separate paper in June 1998.

Code Development, Assessment and Maintenance: With the completion of many of the experimental programs, the main focus of the severe accident research program is currently on the maintenance of those severe accident codes that provide the analytical capability necessary to support the agency’s activities. Because of the difficulty in performing prototypic experiments for a variety of severe accident scenarios, substantial reliance must be placed on these 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 (e.g., steam generator tube integrity issues), to support risk-informed regulatory initiatives or to assess proposed accident management strategies. In the area of severe accident code development, assessment and maintenance, 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, was released in July 1997. Currently, MELCOR is used in conjunction with international cooperative experiments such as PHEBUS and an array of plant analysis associated with specific risk evaluations. Further, MELCOR is one of the most widely-used severe accident codes in the world. As such, RES is continuing the MELCOR Cooperative Assessment Program (MCAP), 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. The release of CONTAIN 2.0, a detailed mechanistic code for the integrated analysis of containment phenomena, was completed in December 1997. With the release of CONTAIN 2.0, the remaining activities for this code involve qualifying the code to perform licensing basis containment analysis calculations, thereby allowing for the replacement of the current licensing codes, CONTEMPT and COMPARE.

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 and was used extensively for analysis in support of NUREG-1570, “Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture.” In addition, SCDAP/RELAP5 was used to support the review of in-vessel coolability and retention of a core melt for the AP600 design. SCDAP/RELAP5 3.2 was released in May 1998. Similar to the MCAP, RES is supporting the SCDAP/RELAP5 Cooperative Assessment Program to promote the assessment of SCDAP/RELAP5 and to provide the NRC with feedback concerning the use of the code by others. Finally, in April 1998, a proposed code strategy was developed that involves the consolidation of a number of the activities associated with maintaining the severe accident codes, including increasing the in-house analytical capabilities to perform analysis previously performed by contractors. This strategy is currently under internal review.

Attachment 2

Accident Management Status

In SECY-97-132, the staff described the status and schedule for licensee implementation of accident management (A/M), plans for evaluating licensee implementation, and the status of the staff's review of the BWROG Emergency Procedure and Severe Accident Guidelines. The staff has made significant progress since then, as described below.

All licensees have committed to implement A/M according to the formal industry position documented in Revision 1 to NEI 91-04, and have provided target dates for completing implementation. Licensee implementation of A/M is progressing. Implementation will be completed at approximately 20 sites by summer 1998, and at the remaining sites by the end of 1998.

At the time of the last status update, NEI had agreed to take the lead in organizing A/M "demonstrations" at four to six plants. These demonstrations would serve the role of the information gathering visits outlined by the staff in SECY 96-088. Following completion of the demonstrations, the staff planned to reassess the need for pilot inspections and the nature of the inspections at the remaining plants. The A/M demonstration phase was completed in March 1998. Four sites were visited -- Comanche Peak (5/97), North Anna (7/97), Duane Arnold (2/98), and Calvert Cliffs (3/98). The demonstrations consisted of licensee presentations concerning site-specific development and implementation activities, followed a tabletop drill demonstrating Severe Accident Management Guidelines (SAMG) use by members of the Emergency Response Organization (ERO) on a second day. Besides NRC staff, each demonstration was attended by 15 to 20 representatives from NEI, other US utilities, and foreign nuclear sites and regulatory authorities.

The demonstrations provided insights into the licensee's A/M implementation and evaluation process. The demonstration plants appear to be addressing the key elements of the formal industry position on A/M, including development of plant-specific SAMG, severe accident training and drills, and administrative controls to maintain A/M capabilities. Also, the SAMG appears to enhance the licensee's drill performance by increasing the ERO awareness of severe accident phenomena and A/M strategy impacts. However, we identified several areas important to effective A/M implementation where we feel additional staff evaluation is needed. These include the integration of SAMG with the Emergency Plan and the Emergency Operating Procedures, the incorporation of IPE and IPEEE insights into the plant-specific SAMG, and the effectiveness of the utility self-assessment process. The A/M demonstration format did not allow sufficient time and flexibility for the staff to evaluate the licensee's supporting assessments in these areas. A further understanding of such aspects of the implementation process is necessary to determine the effectiveness of the voluntary industry initiative and the need for inspections at remaining plants. Accordingly, the staff intends to proceed with the pilot inspections described in SECY-96-088 and SECY-97-132. The objective of the inspections will be to verify that the licensees have evaluated and implemented enhancements to their A/M capabilities according to the formal industry position on A/M, and that these enhancements meet the expectations for A/M set forth in SECY-89-012.

The staff expects to initiate the pilot inspection phase in the summer of 1998, and to enlist NEI's support in identifying utilities that are willing to participate in the inspections. Similar to the A/M demonstrations, the pilot inspections would include review of the licensee's implementation process and observation of a SAM drill, but the inspections would provide additional time and flexibility for staff follow-up on implementation issues and drill observations. A draft Temporary Instruction for use in the pilot inspections was prepared in early 1996. The staff will update the TI to incorporate feedback received on the initial draft and perspectives obtained through the A/M demonstration visits, and will use the revised TI to guide the pilot inspections. Upon completion of the pilots, the staff will establish an approach for confirming the adequacy of A/M implementation at remaining plants and for maintaining oversight of A/M capabilities in the long term. The staff plans to continue its interactions with industry throughout this process, and to consider industry views on the performance criteria and inspection guidance that will be used for judging A/M implementation.

NRC and NEI have been communicating on a number of implementation issues identified by licensees. Additional guidance on these issues was provided in an NEI letter dated July 22, 1997 and an NRC response dated January 28, 1998. In a subsequent letter to NRC (April 3, 1998), NEI expressed concern that the staff may be reversing some previously understood positions regarding A/M implementation, particularly in the areas of licensed operator training and evaluation, use of a systematic approach to training, and application of 10CFR50.59. The staff will provide further clarification on these topics in May 1998. Given this clarification, the staff believes that its views do not represent an escalation of NRC's expectation regarding A/M or a fundamental difference from the NEI guidance. Accordingly, these issues should not be an impediment to the industry completing A/M implementation.

The staff has taken several steps to disseminate information regarding A/M program implementation and products. Emergency preparedness specialists from the region offices participated in each of the A/M demonstration visits to familiarize regional staff on licensee and staff plans. Also, in February 1998 AEOD sponsored oversight training on the Westinghouse Owners Group SAMG for members of the headquarters Reactor Safety Team and staff of the NRC Operations Center. The staff anticipates offering additional training sessions to cover the generic SAMG for other NSSS designs. Finally, NRC has discussed the A/M program and its implications at several emergency preparedness workshops attended by staff from FEMA, the NRC region offices, and state organizations. In addition, the staff plans to make A/M an agenda item for a future NRC-FEMA steering committee meeting.

At the time of the last status update, the Boiling Water Reactor Owners Group (BWROG) had submitted Rev. 0 of the Emergency Procedure and Severe Accident Guidelines (EP/SAG), the staff had identified additional information needs, and the staff was awaiting a BWROG submittal providing this information. The staff received the BWROG response to the request for additional information in January 1998, and received Revision 1 to the EP/SAG document in April 1998.

The staff review has been resumed, and is expected to be completed by mid-1998. This should not affect licensee schedules for completing A/M implementation, since the majority of BWR utilities are working toward a December 1998 completion date.