SECY-96-088

April 29, 1996

FOR: The Commissioners

FROM: James M. Taylor /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:
- SUMMARY:
  - IPE Program:
  - IPEEE Program:
  - Severe Accident Research Program
  - Accident Management Program
- BACKGROUND:
- DISCUSSION:
  - I. IPE Program

PURPOSE:

To inform the Commission annually of the status and progress in implementing the elements of the Integration Plan for the Closure of Severe Accident Issues, i.e., the IPE, IPEEE, Severe Accident Research, and Accident Management programs as requested in a Staff Requirements Memorandum dated April 20, 1989.

SUMMARY:

The Integration Plan for Closure of Severe Accident Issues now has four elements since the Containment Performance Improvement element has been completed and the Commission is being kept informed of the progress in the Improved Plant Operations element by other means (see SECY-95-004).

The summary status of the four elements is as follows:

IPE Program:

A complete report on the status of the IPE program was provided to the Commission

(SECY-96-051) on March 8, 1996. That paper noted that:

(1) Forty-five out of 75 IPE reviews have been completed with staff evaluation reports (SERs) issued. Since that time, seven additional reviews have been completed with SERs to be issued shortly. For those remaining IPEs where the staff can conclude that the licensee met the intent of the generic letter, SERs are scheduled for completion by the end of September 1996.

(2) SECY-96-051 indicated that the staff could not conclude that the licensee met the intent of the generic letter for three IPEs. Since that time, the staff has identified eight additional IPEs of this type. Modified IPEs are being submitted by these licensees to address the staff's concerns. It is expected that these reviews will be completed by the end of December 1996.

(3) Generic insights gleaned from IPEs reviewed along with comparisons with NUREG-1150 results and with the Commission's Safety Goals will be summarized in a NUREG report which will be provided to the Commission in September 1996.

IPEEE Program:

(1) To date, 46 IPEEE submittals have been received with 24 under active review.

(2) To accommodate reductions in funding, the review process for the remaining 51 IPEEE submittals has been revised. This revision reflects the lessons learned in the IPE program, the insights gained from IPEEE reviews now underway, and a recognition that information related to certain generic safety issues, whose resolution is tied to the IPEEE reviews, needs to be included in the reviews.

Severe Accident Research Program

(1) With the issuance of NUREG/CR-6109, "The Probability of Containment Failure by Direct Containment Heating," dated May 1995 and NUREG/CR-6338, "Resolution of the Direct Containment Heating Issue for All Westinghouse Plants with Large Dry Containments or Subatmospheric Containments," dated February 1996, Direct Containment Heating is considered resolved for Westinghouse plants excluding ice condenser plants. Testing has been completed for the Combustion Engineering plants and resolution is ongoing for the remaining PWR Plants.

(2) The Second Steam Explosion Review Group Workshop was held in June 1995 to review the current status of fuel-coolant interaction research, with emphasis on updating our understanding of the α -mode containment failure (containment failure due to an in-vessel steam explosion). A final report of the workshop is in preparation.

(3) Research is continuing in other areas to focus on specific issues, such as debris coolability, hydrogen combustion, and vessel integrity, in order to improve and maintain the NRC capabilities to analyze severe accident issues.

Accident Management Program

(1) The formal industry position on accident management has been accepted by the staff, and licensee efforts to implement accident management programs have been initiated.

(2) Draft inspection guidance has been developed for use by the staff to confirm the adequacy of licensee implementation of accident management initiatives.

(3) The BWR Owner's Group has completed and submitted documents describing their approach to accident management and guidelines for emergency procedure changes related to severe accidents. Staff review of this material is continuing.

BACKGROUND:

On May 28, 1988, the staff presented to the Commission the "Integrated Plan for Closure of Severe Accident Issues" (SECY-88-147). There were six major elements in that plan: Individual Plant Examinations for Internal (IPE) and External (IPEEE) Events, the Severe Accident Research Program (SARP), Accident Management (AM), 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 January 1995 (SECY-95-004).

As noted in SECY-95-004, 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 Accident Management programs.

DISCUSSION:

I. IPE Program

On August 8, 1985, NRC issued a Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants (50 FR 32138) that introduced the Commission's plan to address severe accident issues for existing commercial nuclear power plants. In this policy statement, the Commission addressed its plan to formulate an approach for a systematic safety examination of existing plants to study particular accident vulnerabilities and desirable cost-effective changes. Generic Letter (GL) 88-20, in November 1988, requested all licensees to perform an IPE to identify any plant-specific vulnerabilities to severe accidents, and to report the results to the Commission. Supplement 4 to the generic letter requested licensees to perform an IPE of accidents initiated by external events and also report these results to the Commission.

On May 24, 1993, in an SRM, the Commission requested a paper outlining the major achievements obtained from the IPE and IPEEE programs. It was also requested that the staff include any detailed insights or conclusions on IPE comparisons with the Commission's Safety Goals.

The status of the IPE program was the subject of SECY-96-051. In summary, it indicated that the staff had completed forty-five (out of seventy-five) IPE reviews (SERs issued) and that the preliminary review had been completed on all 75 submittals. Since that Commission paper was written, seven additional SERs have been completed and provided to NRR for transmittal to licensees. It was further indicated in SECY-96-051 that the staff could not conclude, based on the licensees' submittals, that three licensees met the intent of the generic letter. The staff has since identified eight additional submittals of this type. It is expected that the SERs on the remaining IPEs (where the staff has concluded that the licensees met the intent) will be completed by September 1996. Modified IPEs are being submitted by the other licensees to address the staff's concerns. It is expected that the SERs for these revised IPEs will be completed by December 1996.

The IPE Insights Program, established in response to the SRM noted above, is documenting significant insights from the IPE results for the different reactor and containment types. As noted in SECY-96-051, the program is assessing:

- Core damage and containment performance results (e.g., overall core damage frequency, accident sequences, dominant component failure and human error contributors, containment failure modes) relative to the operational and design characteristics of the various reactor and containment types. Methods, data, boundary conditions, and assumptions used in the IPEs have been considered in understanding the differences and similarities observed among the various types of plants.
- Plant improvements identified by the licensees as a result of their IPE efforts and their impact on core damage frequency (CDF) and containment performance.

These insights are being documented in a NUREG report, excerpts of which were attached to SECY-96-051. The staff plans to publish this report for public review and comment in October 1996. The report will be transmitted to the Commission for information in September 1996.

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 initiated by external events and report the results to NRC.

To date, the staff has received forty-six IPEEE submittals, and will receive an additional seventeen submittals in 1996, eleven in 1997, and one with a date not yet determined. Currently, twenty-four 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 analysis, as well as other relevant disciplines. The completion date of the IPEEE submittal reviews, as described in SECY-95-004, was to be the end of calendar year 1998.

As a result of the FY 1996 (and potential future) budget cuts, the staff has revised the IPEEE review process for the remaining fifty-one IPEEE submittals. This revision reflects the lessons learned in the IPE program, perspectives from IPEEE reviews performed to date, and a recognition that information related to certain generic safety issues, whose resolution is connected to the IPEEE reviews, needs to be reviewed. These generic issues are of the following two types:

- 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
- 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 173, "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.

The main elements of the revised IPEEE review process will be:

1. Initial Screening Review

For the remaining fifty-one submittals, the staff will perform screening reviews focusing on: (a) quality and completeness of the submittals and (b) assessments and resolution of the generic issues identified above, including the relevant issues for each plant shown in Table 1. The outcome of this preliminary review will enable the staff to draw conclusions that some submittals have met the intent of the IPEEE generic letter and adequately addressed the relevant generic issues, while others need further review. This review process will be performed by expert contractors with oversight by the senior review board described above. For those submittals falling into the former category, the staff will issue an SER indicating that the intent of GL 88-20 has been met and that relevant generic issues have been adequately addressed.

2. Additional Review of Selected Plants

Additional review is expected to be needed for some IPEEEs which, for example, are poorly documented (thus preventing the staff from reaching conclusions about their adequacy) or have technical deficiencies. These additional reviews will be performed by a combination of staff and contractors, and will also be overseen by the senior review board. An SER will be issued following the completion of the review indicating whether or not the submittal had met the intent of GL 88-20 and adequately addressed the relevant generic issues.

These reviews (both the screening review and more detailed review of selected IPEs) will be performed jointly by RES and NRR and will include the support of contractors.

In addition to the review of the individual IPEEEs, the staff expects to perform some limited study of the set of IPEEEs, like that now underway for IPEs. This study would involve: (1) summary of the significant IPEEE findings and whether any generic observations can be derived, (2) lessons learned about the methodologies used, and (3) assessment of the usefulness of the IPEEE analyses for regulatory applications. This insights study will be undertaken in parallel with the individual reviews, using many of the same contractors.

III. Severe Accident Research Program

The Severe Accident Research Program has continued to 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 addressing the direct containment heating issue, which is expected to be resolved by December 1996. In other areas, such as hydrogen combustion research and fuel-coolant interactions, a limited number of experimental programs will continue to focus on specific issues in these areas and to maintain expertise. The results of the experimental programs are used to develop and validate improved models in NRC's severe accident codes. In many areas, the NRC has participated and will continue to participate, in jointly funded cooperative projects with foreign countries and organizations, in order to leverage our resources. The status of the specific research areas is discussed below.

Direct Containment Heating: Reports on DCH issue resolution for the Zion and Surry nuclear power plants - NUREG/CR-6075 and NUREG/CR-6075 Supplement 1, "The Probability of Containment Failure by Direct Containment Heating in Zion;" and NUREG/CR-6109, "The Probability of Containment Failure by Direct Containment Heating in Surry," were published in December 1994 and May 1995, respectively. The conclusion of these reports is that DCH poses no tangible threat to the integrity of containment for Zion and Surry.

After completion of the Zion and Surry studies, the issue resolution methodology was used to determine if it were possible to extrapolate the plant specific findings to a broad group of similar reactors. NUREG/CR-6338, "Resolution of the Direct Containment Heating Issue for All Westinghouse Plants with Large Dry Containments or Subatmospheric Containments" has been completed, peer reviewed, and was published in February 1996. The results of this work, which addresses 41 plants, show that the conditional containment failure probability, assuming a core melt accident, is less than 0.01. Thus, DCH is considered resolved for all Westinghouse plants, excluding ice condenser plants, and no additional work is required.

Analyses are underway to address DCH in the ice condenser plants. A draft report addressing issue resolution for these designs should be available by August 1996. The remaining large dry reactor containment designs will be addressed in a separate report which factors in the recently completed testing which focused on the phenomena associated with DCH in a CE type design. Large scale integral tests, conducted at Sandia National Laboratories, to investigate DCH in a CE-like design similar to that of Calvert Cliffs were completed in February 1996. The DCH issue resolution methodology, which was previously applied to Westinghouse plants, is now being used for the CE and B&W reactors. A report is scheduled to be available by June 1996.

Fuel/Coolant Interactions: The Second Steam Explosion Review Group Workshop (SERG-2) was held in June 1995 to review the current status of FCI research covering the complete spectrum of interactions, i.e., from mild quenching to very energetic interactions including those that could lead to the α-mode containment failure (containment failure due to an in-vessel steam explosion). The specific objectives of the workshop were to review the status of our understanding to establish a better quantification of the α-mode failure than that determined from the first SERG workshop, held in 1985, and to determine the extent to which our knowledge can be extrapolated to other areas of fuel-coolant interactions. A report on the workshop findings has been prepared and reviewed by the participants. A final report will be published by August 1996.

During this reporting period, the Technical Exchange Arrangement between the NRC and Commission of the European Communities Joint Research Center (JRC), Ispra, was renewed to continue the cooperation in FCI research, in particular, experimental work under the FARO Program. The program itself was extended by the European Communities for another four-year period (1995-98) to cover both in-vessel and ex-vessel FCI as well as ex-vessel melt spreading research. Also, under the previous Arrangement, several steam explosion experiments were carried out at the FARO/KROTOS facility using both prototypic and simulant melt materials. No explosion was observed with prototypic melt even under external triggering, whereas with simulant material (aluminum oxide), a spontaneous explosion was observed.

Other FCI work during this period included the WFCI series of tests (one-dimensional shock tube experiments at the University of Wisconsin) to determine the effect of fuel-coolant mass ratio on the occurrence of steam explosions, and construction of the test facility and testing at the Argonne National Laboratory for the investigation of chemical augmentation of FCI energetics.

Hydrogen Combustion: Under a cooperative program with Japan (MITI/NUPEC), testing was completed for intrinsic detonability and deflagration to detonation transition (DDT) with and without venting in the large scale high temperature combustion facility. This work, conducted at Brookhaven National Laboratory (BNL), is aimed at establishing criteria for detonations of hydrogen-air-steam mixtures at elevated temperatures (500-700K). BNL is currently assessing the DDT results with and without venting. Preliminary assessment of the intrinsic detonability results indicates that the existing analytical model gives reasonable predictions of dominant trends with mixture composition, initial pressure and temperature. Additionally, work at the Russian Research Center, performed under an agreement with the NRC, continued the investigation of hydrogen control issues. Construction of a facility at the California Institute of Technology that will be used to study diffusion flame stability and expansion of high speed jets into hydrogen mixtures has been completed, and the first series of tests injecting hot jets into cold atmospheres is under way. A new research program was initiated at Sandia National Laboratories (SNL) to evaluate the performance of passive autocatalytic recombiners (PARs). Westinghouse proposes PARs in the AP600 design for the control of combustible gases following a design basis accident. The SNL program will be used to perform confirmatory tests to assess PAR performance under a range of hydrogen, air, and steam conditions.

Lower Head Failure/Vessel Integrity: Experiments were performed at Penn State University to study boiling and critical heat flux phenomena (CHF) on downward facing curved surfaces. These scaled experiments are providing part of the data needed to assess whether flooding the cavity (i.e., as an accident management strategy) can remove sufficient heat via ex-vessel cooling to prevent the reactor vessel lower head from failing if large quantities of molten core material relocate to the lower head during a severe accident. Results to date of transient and steady state tests under saturated conditions has revealed that the lowest critical heat flux (CHF) on hemispherical surfaces occurs at the bottom center of the hemisphere (0.4 MW/m^2), and the highest CHF (1.1 MW/m^2) occurs at an angle of 90 degrees from bottom center. A hydrodynamic CHF model for pool boiling on a downward facing heated hemisphere was developed. This model predicts the local CHF for the full size reactor lower head without insulation outside the reactor pressure vessel. Additional experiments are planned to investigate the effect of insulation on the local CHF.

An additional program that will provide valuable information to assess lower head integrity during severe accidents is the OECD RASPLAV project. The formal agreement for this project was signed in August 1994 by the NRC and 12 other OECD member countries. The purpose of this project is to investigate melt-vessel interactions and provide data on the heat flux distribution on the lower head considering the effect of molten ceramic pool natural convection for various melt pool compositions. This project involves large scale integral experiments using UO_2 in representative lower head reactor pressure vessel geometries (i.e., slice geometry), analyses, and related small scale separate effects experiments. Small scale experiments were performed in FY95 and FY96 to demonstrate the feasibility of two different heating methods in the slice geometry (i.e., side wall heating and direct electrical heating) for the planned large scale experiment. Based on results obtained from these experiments, the large scale experiments will be carried out with side wall heating. Tests have also been performed in FY96 in an experimental facility using molten salt as a simulant. In addition, separate effect tests have been performed to measure thermophysical properties of various compositions of corium.

Two new experimental programs on reactor vessel integrity were begun in FY96. The first of a series of eight planned lower head failure experiments was conducted at Sandia National Laboratories in March 1996. The objective of these experiments is to measure strain behavior, time to rupture, and rupture size under the combined effects of various thermal and pressure loads. The second program, being conducted under a cooperative agreement with NRC, EPRI, and three foreign countries, is investigating inherent heat transfer mechanisms that promote in-vessel cooling of core debris and the reactor pressure vessel (RPV) lower head. This is a direct follow-on to the TMI-2 Vessel Investigation Project that was completed in 1994 and addresses the potential for preventing RPV failure during a severe accident with the presence of water in the lower head. Following completion of the reactor vessel integrity experimental programs discussed above, models will be assessed and validated for incorporation into NRC's severe accident codes.

Code Development/Improvement: 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 modified to include abilities to analyze severe accident transients in advanced

LWR plants. With the code modifications completed, demonstration calculations were performed to assess, for significant severe accident sequences, the AP600 plant response. The demonstration calculations were performed to assess the efficacy of provisions for external reactor pressure vessel cooling. Currently, MELCOR is being used in support of the steam generator tube integrity issue, in the evaluation of international standard problems (ISP-37 VANAM-M3), and in conjunction with international experiments (PHEBUS). The ongoing MELCOR Code Assessment Program (MCAP) is an international program to promote the exchange of MELCOR assessment information and to provide the NRC with information concerning the use of the code by others, potential code problems, and innovative modeling techniques.

Significant progress was made on implementing improved models into the SCDAP/RELAP5 code. This code is a detailed mechanistic code for analysis of in-vessel severe accident progression for PWRs and BWRs from the initial phases of an accident, through core degradation and relocation, to reactor vessel or system failure. Changes to the BWR control blade/channel box model for late phase severe accident analysis were incorporated into the code as was an improved model for oxidation of relocating molten material and debris beds. New modeling features unique to the Westinghouse AP600 plant were also added to the code for analysis of severe accidents for this design. In addition, a preliminary design report for improved modeling of late phase debris heatup and melting; molten pool natural circulation and growth; and crust formation, growth, and crust failure was peer reviewed. Implementation of these late phase models was completed. Model validation and code assessment efforts for these late phase models are ongoing. Work is ongoing using the SCDAP/RELAP5 codes in support of the steam generator tube integrity issue.

The latest version of the CONTAIN code was completed and assessed against data from a Westinghouse experimental facility in support of the performance of the AP600 passive containment cooling system (PCCS). In addition, a CONTAIN code user guidance document was generated and the CONTAIN code manual, which documents the latest version of the code, is being updated.

The VICTORIA code, a best estimate fission product behavior code which is used to analyze fission product release and transport in the reactor coolant system, is now undergoing an independent peer review.

IV. Accident Management Program (Lead: NRR)

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-95-004, the staff described the industry commitment to implement accident management pursuant to a formal industry position on A/M, and the remaining activities of the Boiling Water Reactor Owners Group (BWROG) to complete A/M products for BWRs. Significant progress has been made since that time, including acceptance of the formal industry position on A/M, initiation of licensee efforts to implement A/M, development of the approach and draft inspection guidance to be used by the staff for confirming the adequacy of licensee implementation, and completion of the BWROG A/M products.

As described in a January 9, 1995 letter to NEI, the NRC staff has accepted the industry commitment to implement A/M at each NPP pursuant to a formal industry position on A/M in lieu of pursuing other actions for obtaining improvements in industry A/M capabilities, such as issuing a generic letter. The formal industry position calls for each utility to:

> Assess current capabilities to respond to severe accident conditions using Section 5 of NEI 91-04, Revision 1, "Severe Accident Issue Closure Guidelines." and
>
> Implement appropriate improvements identified in the assessment, within the constraints of existing personnel and hardware, on a schedule to be determined by each licensee and communicated to the NRC, but in any event no later than December 31, 1998.

In docketed correspondence to the Commission, all licensees have committed to implement the formal industry position, and have provided target dates for completing A/M implementation. Completion dates range from June 1997 to December 1998, with the majority of the plants completing implementation in the second half of 1998.

Although A/M is being implemented through a regulatory commitment from each licensee on a voluntary basis rather than in response to an explicit regulatory requirement, the staff remains committed to the importance of A/M and to the agency's responsibility to assure a high-quality implementation throughout the industry. To this end, the staff intends 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 include: (1) conducting information gathering visits at 2 to 4 sites in late 1996/early 1997 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 in late 1997/early 1998 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, A/M maintenance will be evaluated on a for-cause basis as a regional initiative. In addition, the staff is evaluating the possibility of participating with licensees in mini-drills or table-top exercises of their severe accident management capabilities and processes. The objective of this participation would be to have a two-way information exchange to enhance NRC and licensee severe accident management understanding, and for the NRC staff to remain cognizant of licensee capabilities in this area.

An initial draft of the TI has been developed and provided to the NRC Regional offices for review and comment and to ACRS for information. Staff plans for inspecting A/M implementation and the major elements of the draft TI were discussed with NEI in a public meeting on February 22, 1996, the ACRS during a subcommittee meeting on March 1, 1996, and NRC Regional office staff during a March 19, 1995 Emergency Preparedness Counterpart meeting. The TI will be updated to incorporate feedback received on the initial draft, as well as perspectives obtained through the information gathering visits.

In SECY-95-004, the staff described the remaining activities of the Boiling Water Reactor Owners Group (BWROG) to complete A/M products for BWRs. (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). The BWROG guidance documents have now been submitted to the NRC. Specifically, an Overview Document describing the BWROG approach to accident management was submitted to NRC on February 3, 1995. A document containing severe accident related changes to the emergency procedure guidelines and additional guidelines for severe accidents was submitted on April 6, 1995.

The staff has initiated a high-level review of the BWR Emergency Procedure and Severe Accident Guidelines but is awaiting BWROG submittal of additional documentation to support this review, including the technical basis document for the EPG/SAG and recent revisions to the draft EPG/SAG. A follow-up meeting to discuss specific staff concerns regarding the BWROG products is tentatively planned for June 1996. The staff anticipates completing review of the BWROG documents by the end of summer. The BWROG plans to finalize these documents following receipt of NRC and utility comments.

The next update on the status of the Severe Accident Integration Plan will be provided in May 1997.

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Attachment: Table 1

ATTACHMENT

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	
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	