March 23, 1984

SECY-84-133

POLICY ISSUE
(Notation Vote)

FOR: The Commissioners

FROM: William J. Dircks
Executive Director for Operations

SUBJECT: INTEGRATED SAFETY ASSESSMENT PROGRAM (ISAP)

PURPOSE: To obtain approval for a pilot program for the Integrated Safety Assessment Program (ISAP), which would evaluate all pending licensing actions and safety issues for an operating reactor.

SUMMARY PROGRAM DESCRIPTION: The Integrated Safety Assessment Program (ISAP) is a new program. It is proposed to be undertaken in lieu of the previously proposed continuation of Systematic Evaluation Program (SEP) and the conduct of the National Reliability Evaluation Program (NREP).

This new approach has evolved from a growing need to provide order and efficiency to the implementation and resolution of licensing requirements for operating nuclear power plants. The benefits of conducting ISAP would be sound regulatory management of the licensing requirements for operating reactors on a plant-specific basis, assurance that the greatest measure of safety is accomplished in the near-term, and the most efficient use of both staff and licensee resources.

The staff proposes that ISAP would be implemented as a trial program consisting of four plants, which would be conducted over the next two years. The licensees for the four plants proposed for the pilot program have volunteered to participate in such a program. The trial program has been constructed such that initial results would be available for the first two plants about a year after initiation to decide whether to continue, redirect or terminate the program. If the NRC elects to continue the program at that time, additional plants could be identified to begin the necessary analysis efforts.

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The features of ISAP, as depicted in Attachment 1 and described in Enclosure 1, are:

1. It is founded on an integrated assessment of all outstanding issues on a given plant:

   a. all previously issued unimplemented licensing actions for the (NUREG-0748) particular plant.

   b. all previously issued TMI (NUREG-0737 and Supplement 1) requirements for the particular plant, yet to be implemented.

   c. the implementation of all resolved generic issues.

   d. the significant safety lessons-learned topics from SEP II.

   e. Unresolved Safety Issues and Generic Safety Issues, to the extent practical.

2. The issues to be evaluated in the program would consider utility-sponsored plant improvements. The review process would ensure that the plant improvements would not automatically become NRC requirements, but would be considered as competing demands for outage time and resources in setting implementation priorities. This aspect is important because the licensees have often argued that their plant enhancements have equal or greater safety significance than some pending NRC requirements.

3. Prior to the start of an ISAP review, each licensee would have an opportunity to request deferral of specific, existing requirements as part of a screening review to establish the initial scope. During the conduct of the program, it would be appropriate to similarly defer implementation of certain new requirements to be addressed in the program.

4. The subsequent integrated assessment process would likely cause some of the deferred NRC requirements to be modified or deleted on a plant-specific basis.

5. The integrated assessment would consider both deterministic findings and insights from a plant-specific probabilistic safety analysis (PSA).

6. The output of the program would be a plant-specific "living schedule" which would be implemented with a license amendment to provide enforceability.

A screening review would serve to identify the plant-specific set of issues to be addressed in ISAP. The staff would evaluate a request by the licensee for deferral of specific implementation requirements and ensure that any other pending license actions are addressed. The results of the screening review would be documented, including the basis for any deferral of specific actions and those implementation requirements that would be completed as scheduled.

In addition, new issues arising late in the review period would be included, except for any for which prompt action is required.

The evaluation tools for ISAP would include: (1) the deterministic reviews of all pending licensing actions and safety issues; (2) a plant-specific Probabilistic Safety Analysis (PSA); and (3) an evaluation of plant operating experience and reliability data, including licensee performance (e.g., from existing SALP evaluations).

The results of these reviews would be evaluated collectively in an integrated assessment. The integrated assessment project team would judge, on balance, the issue-oriented deterministic findings, probabilistic insights and relevant operating experience in relation to plant-specific design features and unique site characteristics. The objective of the integrated assessment would be to identify integral, cost-effective corrective actions to resolve all of the issues raised, including probabilistic-based vulnerabilities, and serve as a technical basis for an efficient plant improvement implementation schedule.

For the pilot program, the scope of the deterministic and probabilistic analyses would necessarily have to be carefully developed to ensure that near-term results could reasonably be achieved. In this regard, the staff expects to take advantage of recently completed SEP reviews and probabilistic analyses (e.g., IREP and high-population density site studies) to provide the working materials for the ISAP pilot program. The staff's approach for the ISAP pilot program would be to review two plants each year for two years, based on requests from licensees to conduct such an integrated review. The two plants to be reviewed in the first year must have an existing probabilistic study which could be readily adapted to the ISAP approach in order to produce results in a year. The second two plants would have a year to develop or upgrade limited-scope PSA to support the integrated assessment process.

If the NRC elects to terminate the program at the end of the first year, the affected licensees would be required to proceed with implementation of the deferred actions. Therefore, the scope of the pilot program should be limited and the licensees should recognize that they proceed with that risk.

The ISAP pilot program should be cognizant of and, as appropriate, would provide an input to other programs investigating probabilistic-based decision processes; e.g., Safety Goal Evaluation Program, Severe Accident Policy, and various Unresolved Safety Issues nearing completion. The staff believes, however, that the ISAP pilot program can and should proceed in parallel with these other programs because the decision process would carefully balance both deterministic and probabilistic judgments on plant-specific basis.

BACKGROUND: The results of the integrated assessments for plants in Phase II of the Systematic Evaluation Program (SEP) were presented in briefing to the Commission on February 4, 1983 and the programmatic experience is described in OPE's evaluation (SECY 83-167).

The conduct of ISAP beyond the pilot program would satisfy Task II.C.2 of the TMI Action Plan for a follow-on program to Interim Reliability Evaluation Program.

OPTIONS: While the staff has considered several options for continuation of SEP and/or NREP and for building upon the lessons learned from SEP Phase II, IREP and other probabilistic analyses, we recommend to forego detailed consideration of these options in favor of a new program.

IMPLEMENTATION: The staff recommends that a pilot program for the Integrated Safety Assessment Program concept be initiated by a Commission Policy Statement and a Staff Requirements Memorandum, as described in the implementation plan in Enclosure 2. The proposed Commission Policy Statement (Enclosure 3) describes the program and would result in some relief for pending operating reactor actions (including resolved unimplemented actions). The Staff Requirements Memorandum should require the staff to inform the Commission of the results of these reviews. It is likely that some issues will require prompt action and these issues would be pursued in accordance with the Commission's recently established backfitting policy.

Enclosure 4 presents a summary of SEP experience which outlines the information requested in the Authorization Bill (HR 2510). The staff will forward separately a formal response to Congress which presents the SEP experience in the context of an approved ISAP plan. The staff will also forward separately a summary of significant findings and experience from prior probabilistic analyses which provides the detailed justification for the Probabilistic Safety Analysis aspect of ISAP.

COSTS: The staff estimates that about 26 PSY and $4000-5000K would be required to complete the review of the four plants under consideration over a two-year period. The current FY 1984-1985 budget for ISAP is 25.2 PSY and $5530K. Therefore, the staff believes that the pilot program can be conducted with the resources budgeted.

A significant resource impact to the staff is the participation of the resident inspector. The resident's contribution is an important feature of ISAP, based on experience from SEP. Although the support by the resident is included in the NRR staff resources allocated to the regions, this staff resource will have to be offset by regional-based personnel to assure that the site inspection functions are maintained.

The cost to the industry for this integrated program, as opposed to continuing the implementation of requirements individually, cannot be estimated at this time. We are continuing to work with the industry to develop resource estimates. It is our belief, however, that the additional resource impact on the industry will not be significantly different than that associated with existing licensing activities. In fact, the licensees who have volunteered to participate in such a program believe that there may be an overall resource savings. The pilot program will provide conclusive data in this regard.

RECOMMENDATION: It is recommended that the Commission:

(1) Approve initiation of the proposed Integrated Safety Assessment Program on a trial basis for four plants; two the first year and two the second year.

(2) Approve the draft Policy Statement in Enclosure 3.

(3) Note that the Commission will be presented with the initial results of the pilot program in about a year after initiation for a decision to continue, redirect or terminate the program.

(4) Note that the Commission will be informed of the results of each review.

SCHEDULING: Commission action should recognize the 60 day review by Congress to release funds previously programmed for SEP Phase III and NREP.

William J. Dircks
Executive Director for Operations

Enclosures:
1. Integrated Safety Assessment Conceptual Description.
2. ISAP Implementation Plan.
3. Proposed Commission Policy Statement.
4. Summary of SEP Experience.
5. Letter from J. J. Ray to Chairman Palladino, dated July 12, 1983.

This paper is tentatively scheduled for discussion at an Open Meeting during the Week of April 2, 1984. Please refer to the appropriate Weekly Commission Schedule, when published, for a specific date and time.

If Commissioners do not vote to approve this paper at the meeting, Commissioners' comments or consent should be provided directly to the Office of the Secretary ASAP thereafter.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT Monday, April 2, 1984, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional time for analytical review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

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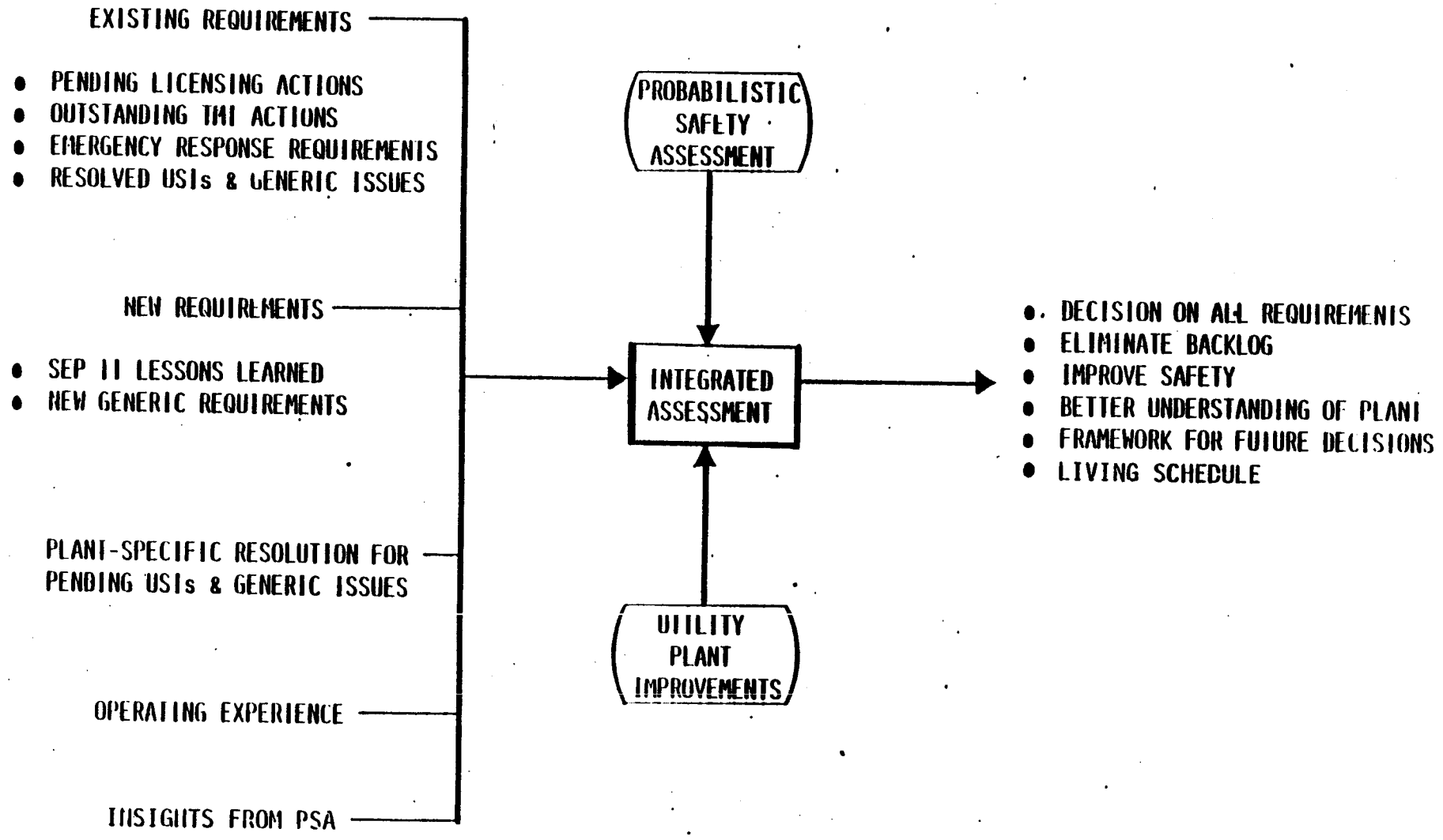
ATTACHMENT 1

INTEGRATED SAFETY ASSESSMENT PROGRAM



ENCLOSURE 1

INTEGRATED SAFETY ASSESSMENT PROGRAM
CONCEPTUAL DESCRIPTION

INTRODUCTION

The objective of an Integrated Plant Safety Assessment Program (ISAP) is to provide a comprehensive review for operating reactors, which will address all safety issues and provide an integrated, cost-effective implementation plan using both deterministic and probabilistic techniques with a dedicated review team that understands the plant design. ISAP would also provide the technical bases to resolve all outstanding licensing actions, establish overall plant improvement schedules and serve as a benchmark from which future regulatory actions can be judged, on a plant-specific basis.

A major attribute of SEP has been the "integrated assessment" process, by which a variety of issues are evaluated collectively on a plant-specific basis. That process serves as the nucleus for ISAP. Other experience from SEP Phase II and experience from IREP and other probabilistic risk assessments have been used to refine that process and estabish a review scope. The following sections describe ISAP in terms of (1) its objective, (2) the review scope, (3) the process, (4) the results, and (5) the resources required.

PURPOSE

The staff and industry have recognized the need for a process to supervise the regulatory requirements for operating reactors. The Commission accomplished this goal for generic issues by establishing the Committee for the Review of Generic Requirements. Similarly, progress is being made on a plant-specific basis with the development of integrated or living implementation schedules.

ISAP takes the next step; it provides a structure for the regulatory management of licensing requirements on a plant-specific basis, assurance that the greatest safety measures will be accomplished in the near-term, and efficient use of both staff and licensee resources. To accomplish this objective, ISAP would (1) evaluate all applicable issues related to plant safety, in accordance with the established scope, as described later, (2) identify cost-effective corrective actions, where necessary, to enhance safety on a plant-specific basis, (3) establish a technical basis to judge implementation schedules, and (4) document the results of the evaluation so that the implementation schedule can be periodically updated, as necessary, to incorporate corrective actions for issues that arise in the future.

The salient aspect of this program is that corrective actions, or the need for any corrective actions, are established on a plant-specific basis. Experience from SEP Phase II and probabilistic analysis reviews indicate that plant-specific features, even between sister plants, have a direct bearing on the relative safety importance of a given issue. Moreover, experience from SEP and probabilistic studies suggests that there are often alternative corrective actions which can provide an equivalent or greater measure of safety, for less cost, when evaluated on a plant-specific rather than generic basis.

SCOPE

The ability to achieve the stated objectives of the program necessarily depends on a clearly defined scope. The review process itself, as described later, will serve to focus the scope for each plant. The issues to be considered in the integrated evaluation should be derived from a specific set of deterministic reviews, a plant-specific "Probabilistic Safety Analysis" (PSA), and plant operating experience. These evaluations would include the following sources:

General Licensing Actions: The Operating Reactors Licensing Actions Summary (NUREG-0748) identifies a specific set of Multi-Plant Actions, TMI Actions, and Plant-Specific Actions for each facility. These lists comprise the pending licensing actions for operating reactors. The TMI Actions and resolved USIs (multi-plant actions) have generic implementation criteria which contain fairly prescripitive requirements. During Phase II, the TMI and USI (multi-plant) actions were deferred to the generic implementation review to avoid a duplication of effort; however, it was apparent that implementation could have been more efficient on a plant-specific basis, because several of the SEP issues were closely related to TMI and USI generic activities. Both the ACRS and staff consultants recommended that implementation of generic requirements should have been addressed in the integrated assessment. Safeguards, quality assurance, emergency planning and waste management issues will be outside the scope of ISAP except under unusual circumstances, such as: (1) the licensee might propose to include such an issue in the ISAP issue list for his plant at his discretion (the staff might or might not agree), or (2) the PSA might flag a particular aspect of QA as particularly important and warranting coverage. In any case, topics will be coordinated with the cognizant program office before any such issue is considered in ISAP.

Reliability Issues: Experience from other probabilistic studies have indicated that there are plant-specific design and operating features which tend to dominate potential plant damage and risk. Plant-specific probabilistic studies have identified specific issues that are not addressed by conventional deterministic reviews. The "Probabilistic Safety Analysis" (PSA) would identify specific potential plant vulnerabilities for dominant accident sequences which would be considered as specific issues in the integrated assessment.

SEP Topics: Experience from SEP Phase II has been used to cull the list of topics reviewed in that phase. Only those topics (review areas) for which corrective action was generally found to be necessary for those plants reviewed in Phase II, and for which significant safety improvements for other operating reactors could be expected, have been retained. The experience the SEP Phase II reviews and the alternative review criteria developed as part of those reviews have been incorporated into a revised set of topic definitions. Because several of the plants in SEP Phase II are fairly representative of the class and vintage of other operating facilities, this set of topics is considered appropriate as a starting point for an ISAP review of the significance of changes in regulatory requirements since the plant was originally licensed.

Plant Improvements: Plant modifications that are developed by the utility staff, which are not licensing requirements, are typically directed at improving plant availability. However, such improvements inherently reduce the potential for plant transients and, therefore, reduce challenges to safey systems. In addition, there are some utility-initiated actions that directly affect safety. Consequently, at the request of the utility, such actions should be considered in an integrated assessment to ensure that they receive appropriate consideration in the scheduling and limited resources for overall plant improvements. The staff would implement administrative controls to ensure that utility plant improvements do not automatically become licensing requirements, in a manner consistent with the backfitting policy.

Operating Experience: In SEP Phase II, reviews were conducted of plant operating experience by compiling and evaluating availability/capacity factors and operating events (forced shutdowns and reportable occurrences). In addition, the Regional staff provided a synopsis of the Systematic Assessment of Licensee Performance (SALP) findings for the specific plant under review.

These reviews proved to be useful in providing another level of perspective for the integrated assessment. However, issues raised in these reviews were only addressed as they applied to specific issues raised in the topic evaluations (e.g., Palisades management performance for procedural controls and Millstone 1 gas turbine-generator failures). The staff has concluded that the operating experience evaluation is equally significant to an assessment of overall plant safety as the issues raised in topic evaluations or a risk analysis. This evaluation would serve to (1) confirm the adequacy of the data used in the plant-specific PSA and (2) identify strengths and weaknesses in the plant operating and emergency procedures and maintenance practices that should be considered in judging the adequacy of proposed corrective actions to resolve an issue or set of issues. As described in the evaluation process below, the staff has concluded that an ISAP review team should include a human-factors specialist, in order to ensure that procedural and maintenance issues receive appropriate consideration.

Unresolved Generic Issues: Those Unresolved Safety Issues (USIs) and other generic issues for which staff-approved implementation criteria exist are multi-plant actions and, therefore, fall under the category of General Licensing Actions. However, there are a number of these issues for which implementation criteria have not yet been developed. In some cases, the unresolved issue is well enough defined or sufficient research has progressed such that the issue could be resolved in an integrated assessment on a plant-specific basis without implementation criteria. Because of the uncertainties in the status and applicability of specific unresolved generic issues relative to this process, the staff believes that these issues should only be addressed as explicit issues where the staff and licensee agree that consideration in the integrated assessment would likely resolve the issue. Otherwise, these USIs and generic issues would only be addressed in the context of the PSA, as described later.

EVALUATION PROCESS

The evaluation process for ISAP has been developed principally from the programmatic lessons learned from SEP Phase II. The evaluation process for the PSA has been developed from experience with the Indian Point, Zion, Big Rock Point, and Limerick PRAs. The process described below is for an "equilibrium" evaluation; i.e., a plant review conducted when the program is well underway and beyond the startup effects. For the pilot program, the major steps in the process will be the same, but the specific evaluation process would have to be modified so that the staff could provide near-term results to establish the viability of the program.

An important aspect of this program is that, at the time a decision is made to begin an ISAP review, all outstanding licensing requirements for which implementation is not complete (i.e., design, procurement, and installation) would be screened for consideration in a composite evaluation of issues, including those actions for which resolution has been defined but not yet implemented. At the licensee's request and with adequate justification, specific implementation requirements would be deferred for evaluation in an ISAP review.

An important programmatic lesson learned from SEP was that an integrated assessment can only be successfully completed in a stable environment. Thus, any new licensing requirements that may evolve during the course of or following ISAP, for which the Commission or other appropriate authority does not require prompt action, would be deferred to ISAP or some later update of the ISAP results.

In order to ensure a consistent and effective review, the staff would prepare ISAP procedure guidelines that describe the functional steps and decision process, following approval of the program concept. These guidelines would establish the detailed review procedures and the management review and approval process.

Screening Review: The first step in ISAP would be a meeting, or a series of meetings, between the staff and licensee to review the scope and define the set of review areas to be addressed in a plant-specific evaluation. Where the staff and licensee agree that specific review areas do not constitute significant safety issues for plant-specific reasons, those review areas would be excluded from further consideration to enhance efficient use of resources. The bases for excluding specific review areas would be documented and incorporated in the draft and final reports.

If a determination were made that implementation should proceed for specific actions because of safety considerations, they would not be included in ISAP and would proceed on their assigned schedule. Implementation requirements deferred to ISAP would have a documented justification for deferral. The implementation schedule for the results of ISAP would be based on the importance to risk as developed from deterministic and probabilistic reviews.

The screening review would apply to the SEP Topics, General Licensing Actions, and Unresolved Generic Issues. The resultant list of review areas are hereafter referred to as ISAP Topics.

At that time, the procedures for the conduct and identification of issues for the plant-specific PSA and Operating Experience evaluation would be established. These evaluations would be conducted in parallel with the ISAP Topic evaluations, as described below, such that all of the issues would be defined and their interrelationships established prior to the start of the integrated assessment. At the end of the screening review, schedules would be established by which the licensee would prepare and submit safety analyses for each ISAP Topic, unless an analysis already exists. These schedules would be used to monitor program progress, allocate staff resources, and coordinate integrated assessment schedules.

ISAP Topic Evaluations: The ISAP Topic evaluations would be conducted in a manner similar to that used for the SEP Phase II Topic evaluations. The licensee would prepare a safety analysis for each ISAP Topic, comparing the facility design to the prevailing review criteria. For the SEP Topics, review criteria and guidelines already exist. For other licensing-related topics, the generic implementation criteria or Standard Review Plan acceptance criteria would be used. If criteria other than current SRP requirements are used, they will be carefully developed and documented. The safety analyses would be prepared in a standard format, identify variations from the acceptance criteria, describe any mitigating features of the plant design or other rationale by which the variance should be judged and, subsequently, be submitted for staff review. In some cases (particularly for General Licensing Actions), such analyses may already exist and may be incorporated by reference or updated as necessary. The staff would review the licensees' safety analyses and prepare individual safety evaluation reports (SERs) which identify differences from the review criteria as issues to be addressed in the integrated assessment. In some cases, the staff evaluation may conclude that the plant provides a measure of safety "equivalence" as a result of the mitigating features of the plant design or on some other defined basis. The resulting set of plant-specific SERs would serve to identify the deterministically-based ISAP Topic issues.

In the event that either the licensee or the staff should determine during the course of a topic review that a difference warrants prompt action to protect the health and safety of the public, that action will be pursued expeditiously and independent of ISAP.

Probabilistic Safety Assessment: As part of ISAP, the staff would request that the licensee conduct a plant-specific PSA, in accordance with the Procedures Guide, NUREG/CR-2815, and submit the results for staff review. In addition, the staff would identify the manner by which unresolved generic issues should be considered in the PSA (see Scope, Unresolved Generic Issues), with particular attention to the generic issues ranked high in NUREG-0933. The licensee will use the results of the PSA to identify plant design and operation weaknesses that warrant consideration in the integrated assessment, address the generic issues, and provide specific risk perspectives for the deterministic topic evaluations.

As described in Enclosure 2, the staff would rely on volunteering utilities for the pilot program. Therefore, the scope of the PSA for the pilot program would have to be established on a case-by-case basis to ensure an achievable schedule. A final version of a general PSA procedures guide, including external events, is scheduled to be completed in September 1984.

As part of the staff review, the "back-end" portion of the PSA (i.e., containment failure characteristics, radionuclide releases, and resulting consequences) would be established for the significant accident sequences. The staff does not believe that the "back-end" analysis techniques are sufficiently developed to require that they be performed by the licensees, at this time. Consequence evaluations have been performed by Sandia National Laboratories using the stylized "Siting Source Terms" (NUREG/CR-2239). Most accident sequences can be associated with source terms using interpolation among published studies and these, in turn, can be associated with Siting Source Terms. In this way, consequences can be roughly estimated for each accident sequence. The Severe Accident Research Program is also addressing the "back-end" portion of a probabilistic analysis. The generic aspect associated with containment failure probabilistic and source terms would be presented in a SARP reference document. These generic data will be incorporated into the staff assessment of cost/benefit when the reference document is issued.

The staff believes the savings in licensee costs associated with completing their PSA studies in this way warrants the lesser accuracy of the suggested methods, at least for the interim until containment analysis codes and procedures and source term estimation techniques can be developed which are sufficiently practical and efficient to include in the procedures guide.

Based on the review of the licensee's PSA, the staff would prepare a safety evaluation which identifies relative strengths and weaknesses in the plant design and operation and confirms the licensee's risk perspective for the topic evaluations. The PSA evaluation will provide a measure for estimating the potential impact of modifications associated with the issues raised in the topic evaluations, where amenable, as well as identify dominant contributors to risk and core melt as issues to be considered in the integrated assessment.

Operating Experience: In SEP Phase II, plant-specific evaluations were conducted on the causes of forced reactor shutdowns and the causes and categories of reportable events. Both the ACRS and staff's consultants commented that these operating experience evaluations provided valuable insights into the plant design and operation. The staff continues to believe that, consistent with an objective to assess overall plant safety, the operating experience evaluations should be performed. General strengths and weaknesses in plant operation and maintenance experience should be explictly addressed, as they relate to proposed corrective actions for other issues.

The operating experience evaluation would consist of a collection and evaluation of forced shutdowns and reportable events. Operating experience from the Nuclear Plant Reliability Data System would also be included. The events would

be categorized in accordance with a judgment of the related safety significance by determining cause and effect through an interaction with the plant staff and Regional personnel. The resulting report would (1) summarize the events in terms of the significance categories, like those prepared for SEP Phase II; (2) identify common-cause effects, design deficiencies, and the positive and negative aspects of management, operation and maintenance practices; and (3) summarize overall licensee performance (e.g., as indicated by SALP and INPO evaluations).

INTEGRATED ASSESSMENT

The topic evaluations, PSA review and operating experience evaluation will result in a specific set of issues to be addressed in an integrated assessment. This list of issues would be forwarded to the licensee to prepare a proposal for staff consideration. The objective of the integrated assessment would be to determine which issues warrant corrective action, based on perceived safety significance, and to develop cost-effective corrective actions which will resolve multiple issues, where practical, to enhance the overall safety of the plant.

In parallel with the licensee's integrated evaluation, a staff dedicated review "team" would review the issues and familiarize themselves with the plant design. The team would tentatively consist of an Integrated Assessment Project Manager (IAPM), the Operating Reactor Project Manager (ORPM), the Resident Inspector (RI), a Reliability Analyst (RA), and a Human Factors Specialist (HFS). This is essentially the same kind of review team that was used for SEP Phase II, except that the human factors specialist has been added to provide a better perspective of management, procedures and maintenance practices. The review team would meet at the plant site, perhaps several times, to go over each of the issues "in-situ," to ensure that each issue is clearly defined and any pertinent plant-specific design or operation features are clearly understood. The review team would also draw on specialized technical expertise from within staff to understand the safety ramifications of each issue and pertinent research data or general operating experience that should be considered.

Following the licensee's submittal of an integrated assessment proposal, the review team would evaluate that proposal by means of a collective judgment of the significance of the differences and the appropriateness of the proposed corrective actions. The judgment would be based on both the deterministic and probabilistic information developed during the course of the review using the guidelines contained in a review procedures manual. The staff decisions on requirements for ISAP subjects are intended to resolve all pending license actions, assure the reactors pose no undue risk, and achieve full compliance with the reactor safety criteria at minimum cost and with minimum delay. The decision criteria to be employed by the staff would entail weighing the following considerations: (1) compliance with rules, (2) compliance with regulatory guidance, (3) effectiveness of minimizing offsite radiological risk, as indicated by probabilistic safety analysis, (4) speed of implementation, (5) effects of a decision on defense-in-depth, and (6) cost/benefit considerations.

Departures from literal compliance with regulatory guidance would be considered when (1) the safety objectives can be met in a less costly or more expeditious way through alternatives, or can be met in a way that better strengthens defense-in-depth, (2) the probabilistic safety analysis might suggest that more stringent requirements may be appropriate to deal with dominant contributors to risk, and (3) when it can be shown unambiguously that the safety issue is not significant to offsite radiological risk at the facility. When regulations or license conditions are not explicitly complied with, exemption requests will be processed in accordance with established procedures.

The probablistic safety analysis will be used primiarly to obtain perspectives on the ways the many licensing issues contribute to the safety of the plant and to identify dominant contributors to the risk. The staff does not propose to employ the quantitative design guidelines in the proposed Commission safety goals as decision criteria, consistent with the Commission policy on interim use of the safety goals and the larger uncertainties in bottom-line risk predictions. A partial exception is the use of $1,000 per person-rem averted to enable benefit/cost evaluations to be made of hypothetical alterations in plant design and/or operation. Such calculations can be made more accurately with probabilistic analysis techniques then can absolute risk predictions.

At the licensee's request the staff may also consider averted economic losses to the licensee in comparisons of prevention versus mitigation strategies, although it is not the staff's intent to order plant modifications not warranted by public health and safety considerations.

Great care will be taken to consider the uncertainties in the probabilistic safety analysis. Each important inference for regulatory action to be drawn from the PSA will be treated as an hypothesis to be examined against all available evidence, rather than as a fact. Since probabilistic safety models are a collection of simple assumptions in a coherent framework, an assessment of uncertainties will be necessary for the significant insights.

The integrated assessment will also consider the status of pending Commission policy issues such as safety goal policy, severe accident policy, backfitting policy, source term revisions and living schedule. These will be phased into the program as evaluation criteria evolve which can be reasonably assessed on a plant-specific basis. In the interim, however, the ISAP program will provide a cost-effective basis for tradeoffs on improvements and scheduling that will enhance the public health and safety. Any affected ISAP decisions made prior to the resolution of these programs could be reassessed by the licensee or the staff soon after the Commission policy is finalized. With regard to safety goal policy, ISAP does not violate the admonition not to use safety goals during the evaluation period. With regard to severe accident policy, it looks increasingly likely that any implementation plan for severe accident issue resolution on operating reactors will require an overall plant evaluation resembling ISAP. This adds to the incentive to run the ISAP pilot program to gain experience with the concepts involved. ISAP will adhere to administrative procedures for backfitting. It provides a mechanism to be more discriminating in ordering backfits, and to use staff-licensee consensus whenever possible. The purpose of staff-performed containment/source term

calculations in the pilot ISAP cases is to utilize the latest state-of-the art from the Accident Source Term Project Office, as we proposed to do in the ongoing standards development contexts. Once standards are set for such analyses in the future, ISAP is thought to be the best way to implement the concept of living schedules, and is fully compatible with the concept. Should the living schedule conception evolve in different directions, we would expect the ongoing ISAP pilots to follow suit.

The IAPM would formulate proposed staff positions from the review team's judgments, in the form of a draft report. The draft report would contain a summary of the plant description and review process, incorporate the pertinent background material by reference, and present a detailed description of each issue with the associated staff position and a discussion of the basis for that position. Following division-level comment and concurrence, the draft report would be published for comment. The staff believes that the review process developed as part of SEP Phase II should be continued; i.e., ACRS review and an additional peer review by a panel of recognized experts in broad areas of nuclear safety.

RESULTS

The draft report would serve as a vehicle to resolve all of the outstanding issues. In the event that the staff and licensee disagree on the resolution of any given issue, that difference would be aired during the public comment and peer review period, resolved in accordance with established appeal procedures, and ultimately presented to the Commission as part of the final report.

The licensee would be requested to propose and justify an integrated implementation schedule to be included in the final report. Any disagreement between the staff and licensee on schedules would follow the same course as described above.

The final report would serve to support a license amendment incorporating the integrated schedule, providing enforceability of the results. As new issues arise in the future, for which prompt action is not required, the resulting implementation schedules would be deferred and the integrated schedule would be periodically upgraded to incorporate those actions. The staff believes that the updating interval should be approximately three to four refueling outages (but no more than five years), as described in the implementation plan in Enclosure 2. Alternate updating intervals might be acceptable; the optimum interval depends on the detailed procedure for updating, must ensure that the integrated schedule is not unnecessarily disrupted, and consider the relative safety significance of each action involved.

At the conclusion of ISAP, the licensee and staff would have the benefit of a plant-specific PSA. The licensee could use the PSA to improve procedures, maintenance and training programs, evaluate operational occurrences, and optimize the design of plant improvements. The staff would have the benefit of new information to add to general probabilistic analysis experience and a basis from which to judge the need for prompt action on future issues on a plant-specific basis.

As to enforcement bases, ISAP decisions will have the same effect that backfitting decisions have for generic issues, though it would likely yield greater plant-to-plant variability. DRA/RES is currently working with IE on prioritization of inspection modules based upon risk. Better plant specific identification of the design and operation characteristics most significant to safety would facilitate improved risk-relevance in inspections, and is thought to be a potential benefit of ISAP, albeit at some cost to standardization of inspection priorities.

RESOURCES

The resources required to conduct an ISAP are reflected in the proposed NRR budget projections and consider experience from SEP Phase II and probabilistic studies and consultation with experienced industry firms and utilities.

NRC Staff: The staff resources to conduct a full-scope ISAP review are estimated to be 9.2 PSY and $1.4 to $1.7 million per plant. For the pilot program, the staff estimates that the resources required to conduct the review, utilizing existing deterministic and probabilistic studies, would be 10.9 PSY and $1600K in the first year and 14.9 PSY and $2400K in the second year; this compares reasonably well with the budgeted resources of 11.4 PSY an $1330K in FY 1984 and 13.8 PSY and $4200K in FY 1985.

The resource estimates for the topic evaluations and integrated assessment were derived from SEP Phase II resource data on a per topic basis, averaged over all of the plants reviewed. Because of the differences in the manner by which topic evaluations were conducted in Phase II from the approach proposed for ISAP (i.e., more licensee involvement) and the larger scope of topics in ISAP, the staff believes that ISAP plant reviews could be conducted within the projected budget by adjusting the pace as experience is gained in the beginning of the program. In addition, the majority of the staff resources for the review of non-SEP topics are included in the projections for operating reactor actions.

An estimate of the staff resources required to conduct the PSA review have been developed based on experience, particularly Indian Point and Zion. The PSA schedules would be based on intermediate reports by the licensee so that the staff can track the progress of the analysis, review results as they are completed, and interact with the analysis team to understand the plant design and analysis assumptions. The staff review would concentrate on omissions, modeling approximations, marginal subjective judgments, data utilization, success/failure assumptions, and common-cause failures.

Industry: The resources required by the utility to conduct an ISAP are estimated to be as much as $5,000,000 for the topic evaluations, integrated assessment, and plant-specific PSA. These resources reflect both utility staff and contractors, assuming $100,000 per person-year of effort. The estimate for the topic evaluations and integrated assessment was derived from the average utility costs to conduct a Phase II. Like the staff resources, the staff expects that a majority of the non-SEP topic would have already been done or at least accounted for in some other way.

However, this esimate does not consider the difference in utility resources required to resolve issues independently as opposed to an integrated resolution of all issues. It is concievable that the resources required to resolve these issues independently is greater that the resources required for ISAP.

The resources required to conduct a plant-specific PSA have been developed from the IEEE/ANS procedures guide, projections for an industry study of Oconee, and conversations with PRA analysts. The industry expects that there are sufficient experienced analysts available to support between six to twelve studies per year.

ENCLOSURE 2

ISAP IMPLEMENTATION PLAN

INTRODUCTION

The staff recommends that an Integrated Safety Assessment Program (ISAP) should be implemented by a Commission Policy Statement, with an accompanying Staff Requirements Memorandum to commence the program. The staff considered alternative ways to implement this program (e.g., rulemaking); however, because of the extended scope of this program, the need for a trial period, and the complex interrelationships with other regulatory activities (e.g., backfitting, severe accident policy, and safety goal), the staff concluded that a policy statement would be the most appropriate vehicle. The staff believes that ISAP could be conducted within the existing regulatory framework; however, the staff recognizes the merits of formalizing the program with rulemaking. Therefore, the staff will consider rulemaking for ISAP during its initial phases, as experience is gained to demonstrate the important aspects of the approach. The proposed policy statement is presented in Enclosure 3.

Inasmuch as ISAP is a fairly substantial change in the way that operating reactor actions have been dealt with in the past, and in consideration of the time required to conduct an ISAP evaluation, the staff believes that the program should initially be conducted as a pilot study. The following sections describe the schedules and the associated rationale, the bases for plant selection, and the method for applying ISAP results.

PHASED SCHEDULES

The staff proposes that ISAP be implemented with a pilot program involving four plants. The review schedules would be established so that two plants would be completed in about a year.

In order to provide near-term results to assess the efficacy of the program, the pilot program must rely heavily on existing deterministic and probabilistic analyses; e.g., SEP, IREP, and high-population density site studies. Because the PSA is the more time consuming and resource intensive, the first two plants must have existing plant-specific probabilistic studies that can be readily applied to develop risk insights for the integrated assessment. Any additional deterministic reviews could be phased in parallel such that all of the issues for the first two plants are well defined in about nine months, allowing three months to conduct the integrated assessment.

The two plants to be reviewed in the second year would have about a year to develop or upgrade a plant-specific PSA. Similarly, any necessary deterministic reviews could be conducted in phases in parallel. This second group of plants for the pilot program provides (1) experience in the conduct of the supporting analyses, (2) additional experience for the procedures and resource estimates for the program, and (3) continuity for any continuation of the program. This experience would enhance the decision on whether the program should be continued or redirected. If the NRC elects to terminate the program based on the results of the first group of plants, the analyses conducted to that point for the second group would still contribute to the resolution of the issues involved; however, the affected licensees would be expected to promptly begin implementation of the actions deferred.

PLANT SELECTION CRITERIA

For the purpose of a pilot study, the staff intends to select and schedule plants from the licensees who have expressed an interest to voluntarily participate in such a program. These plants would be scheduled based solely on the availability of probabilistic and deterministic analyses consistent with the resources designated for the program.

The staff expects that other licensees may be willing to voluntarily participate in ISAP with the understanding that many pending licensing requirements would be evaluated in a plant-specific integrated assessment. Under such circumstances, the licensee could redirect resources to an ISAP evaluation from the individual licensing action evaluations.

RESULTS APPLICATION

The final ISAP report for a given plant would specify implementation schedules for any necessary plant improvements, including procedural and Technical Specification changes. The evaluation would serve as the basis by which the staff would conclude whether the schedules proposed by the licensee are appropriate. The final report would then be referenced by a licensee amendment formalizing the implementation schedules.

During, and following, an ISAP, new licensing issues will undoubtedly arise which will have to be addressed and, where necessary, incorporated into the implementation plan. Such new licensing issues for an ISAP plant would be considered for deferral to a subsequent periodic update of the implementation schedule. Those issues for which quicker action is required for the health and safety of the public would be appropriately pursued in accordance with the established procedures for backfitting.

The staff expects that the implementation schedules resulting from ISAP will cover several plant outages. A reasonable interval for periodically updating the ISAP to evaluate the deferred issues would be about three to four refueling outages, but not more than five years, and the provision and schedule for the update would be included in the license amendment. This approach is generally consistent with a conceptual "Five Year Backfitting Plan" discussed with the Commission and subsequently submitted in a letter dated March 17, 1983, by Combustion Engineering which was forwarded to the Commission by a memorandum dated May 2, 1983.

An update evaluation would generally be conducted like an ISAP evaluation, except that the scope would be limited to the deferred issues and the licensee's proposed actions would be related back to the final ISAP report. Following a staff review of the proposed update, to be coordinated by the Operating Reactors Project Manager, the staff would issue a new license amendment, which specifies the schedule for the next update, and the licensee would update the plant-specific PSA. This process would continue for the

remainder of the life of the plant. Inasmuch as the scope of the update evaluation would be limited to new issues and the initial implementation schedules would be issued on a staggered basis, the staff believes that the resources required to conduct the update reviews would be within those projected for operating reactor licensing actions, because the reviews would be more efficient.

ENCLOSURE 3

NUCLEAR REGULATORY COMMISSION
10 CFR PART 50

Commission Policy Statement on the
Systematic Safety Evaluation of Operating
Power Reactors

AGENCY: Nuclear Regulatory Commission

ACTION: Notice of Commission Policy Statement

SUMMARY: This Policy Statement describes a pilot study for a change to the regulatory practices regarding the evaluation of safety issues for operating nuclear power reactors. The approach to be used will address significant regulatory requirements which have evolved since the plant was originally licensed and pending licensing actions which have evolved from a variety of other sources. An integrated assessment will be conducted on a plant-specific basis, as part of a trial program, to evaluate all licensing issues on a given facility and to establish schedules for any necessary plant improvements. In addition, procedures have been established to allow for a periodic updating of the resulting implementation schedules for new licensing issues that arise in the future.

FOR FURTHER INFORMATION CONTACT: Frank J. Miraglia, Assistant Director for Safety Assessment, Division of Licensing, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Telephone (301) 492-7493.

SUPPLEMENTARY INFORMATION: In 1977, the Nuclear Regulatory Commission initiated the Systematic Evaluation Program (SEP). Phase I of SEP defined a specific set of safety issues (topics) to be reviewed for operating nuclear power plants. Phase II of SEP was a pilot review of those topics for eleven of the oldest domestic operating reactors. Results have evolved from SEP over the last two years and identified significant experience relative to the safety and evaluation techniques for operating plants.

In 1980, Congress enacted Public Law 96-295 (the NRC Authorization Bill for Fiscal Year 1980). Section 110 of P.L. 96-295 required that the NRC develop a program for the systematic safety evaluation of operating reactors. The program proposal would have extended SEP to an evaluation which required licensees to compare their plant design to the acceptance criteria in the Standard Review Plan (NUREG-0800), That program was not implemented for operating reactors; the Commission determined, and the Congress agreed, that the scope of the program was too broad to efficiently evaluate the safety of operating reactors. Congress subsequently specified in later Authorization Bills that funds should not be spent to implement that program. However, those activities were useful in that they focused attention on the needs and difficulties associated with the systematic safety evaluation of operating reactors as they relate to a constantly changing technology and increasing scope of regulatory requirements.

Following the TMI-2 accident, the NRC developed the TMI Action Plan (NUREG-0660) from the safety lessons learned. Two aspects of the TMI Action Plan are particularly significant to the evaluation of the safety of operating plants: (1) it identified a large number of corrective actions to be implemented by operating plants and (2) it initiated the Interim Reliability Evaluation Program (IREP) in which plant-specific probabilistic risk assessment (PRA) studies were to be performed by the staff for several operating reactors to supplement the risk-reliability experience from the Reactor Safety Study (WASH-1400). The licensing actions resulting from TMI have increased the scope of outstanding licensing issues for all operating plants. Similarly, the experience thus far from IREP indicates that there are plant-specific strengths and weaknesses, from a reliability point of view, that warrant further consideration, beyond the deterministically-based issues.

One of the most significant conclusions drawn from SEP and IREP is that issues related to safety of operating nuclear power plants can be more effectively and efficiently implemented in an integrated, plant-specific review. In addition, the experience from SEP has served to focus on the set of current licensing criteria which should be evaluated for operating plants and experience from IREP has served to define the methods to conduct a plant-specific probabilistic safety analyses so that consistent, comparable results could be obtained which would enhance an integrated plant safety assessment.

Historically, licensing issues have been evaluated generically and guidelines for any necessary corrective actions have been applied uniformly to all plants. While this approach has provided an effective means to ensure resolution of these issues, the generic implementation has not given sufficient attention to plant-specific characteristics which have a direct bearing on the appropriateness of the corrective action and the relative importance of the issue in relation to an overall plan for any necessary plant improvements. In some cases, consideration of plant-specific characteristics have identified alternate corrective actions which provide an equivalent or greater measure of safety, often at less cost to the licensee.

Consequently, the NRC has developed the regulatory procedures and attendant policies to conduct integrated assessments for operating power reactors. This approach is called the Integrated Safety Assessment Program (ISAP). In order to ensure the effectiveness of this program, it will be started on a trial basis and the plants to be reviewed have been selected by the NRC Staff from among those licensees who indicated an interest to voluntarily participate in such a program.

Based on the results of this trial program, the NRC will decide, in about a year, whether or how this program should be extended to other operating reactors.

IMPLEMENTATION SCHEDULES

To provide a stable environment to conduct ISAP, the Commission has authorized the staff to suspend specific existing implementation schedule requirements for the plants to be reviewed. Each affected licensee will be expected to propose and justify deferral for specific implementation requirements that warrant further evaluation. The associated implementation requirements and other safety issues will be evaluated collectively in an integrated assessment. The staff is only authorized to defer substantive regulatory and other requirements to the extent allowed by the Commission's procedural regulations. Thus, the staff will use the provisions in Section 50.12 to grant any exemptions.

In addition, any new implementation requirements which evolve late in the course of or following ISAP will be deferred for the plants involved and incorporated in an implementation schedule update, as described below.

The only exceptions will be issues for which the NRC Staff explicitly determines that prompt action is required to protect the health and safety of the public, and that the action is necessary in accordance with 10 CFR 50.109. Such actions include the short-term response to bulletins issued by the Office of Inspection and Enforcement.

SCOPE OF EVALUATION

The scope of ISAP is intended to be as comprehensive as practical. Consequently, it will consist of deterministic, probabilistic and operating expeience evaluations, which will serve to identify specific issues to be addressed in an integrated assessment.

The deterministic review areas, or ISAP topics, will be derived on a plant-specific basis during a screening review with the licensee at the beginning of the program. The issues to be considered are (1) a set of SEP Topics for which the NRC Staff has found significant differences between current licensing criteria and typical design criteria in existance when operating plants were licensed; (2) all pending licensing actions for the plant, including multi-plant actions, TMI Action Plan requirements and plant-specific licensing actions; (3) the unresolved generic issues for which resolution on a plant-specific basis might be expected, and (4) plant improvements proposed by the licensee.

The Commission's Safety Goal Policy published on March 4, 1983, (48 FR 10772), indicates that the quantitative goals and design objectives will not be used in the licensing process during the evaluation period, nor will the policy be interpreted as requiring that licensees or applicants perform a probabilistic analysis; however, the Commission continues to believe that probabilistic analyses provide a valuable adjunct to the deterministic regulatory requirements and enhance engineering judgments, if they are properly performed and applied. Consequently, the Commission believes that a plant-specific probabilistic safety assessment (PSA) should be performed in conjunction with

ISAP. The plant-specific PSA will provide a basis for cost/benefit evaluations for the deterministically-based issues and will also identify potential strengths and weaknesses in the plant design and operation which should be considered in an integrated assessment.

An operating experience evaluation will be conducted in parallel with the topic evaluations and plant-specific PSA. This evaluation will be used to identify issues related to significant trends, event precursors, plant management and operation, and maintenance practices. In addition, the operating experience evaluation will provide a diverse perspective for the integrated assessment. The evaluation will consist of an analysis and categorization of reportable events and forced plant shutdowns and an evaluation of overall licensee performance.

EVALUATION PROCESS

The ISAP Topic evaluations and plant-specific PSA will be conducted in parallel. The licensee will initially perform deterministic analyses for the plant-specific set of ISAP Topics by comparing the as-built design of the facility to the current licensing criteria, industry codes and standards, or other appropriate acceptance criteria and also provide risk perspectives for each issue based on the PSA. The staff will review the licensee's analyses and issue safety evaluation reports which identify specific differences from the acceptance criteria and any attendant safety issues which should be considered in the integrated assessment. Schedules for the licensee's analyses and staff evaluation will be established during the screening review to enhance an efficient use of resources.

A PSA would be conducted by the licensee in accordance with an NRC Procedures Guide (NUREG/CR-2815). The Procedures Guide describes appropriate and consistent methods for (1) initiator definition, (2) application of data, (3) success/failure criteria, (4) analysis, (5) quality control, and (6) documentation and presentation of results. In addition, the NRC Staff will identify methods by which the PSA should address unresolved generic issues; i.e., safety issues for which acceptance criteria do not yet exist. During the screening review, milestones will be established to monitor the progress of the PSA and to ensure appropriate interaction between the NRC Staff and the licensee. The licensee will be expected to use the PSA to identify dominant contributors to risk that should be specifically considered in the integrated assessment. For the trial program, the extent and nature of plant-specific probabilistic analyses will be established on a case-by-case basis.

The issues raised in the ISAP Topic evaluations and the PSA, and the operating experience evaluation will be considered collectively in an integrated assessment. The NRC Staff will present its conclusions regarding the need for or appropriateness of corrective actions proposed by the licensee for each of the identified issues in a draft report. The draft report will be issued for public comment and peer review. Should the NRC Staff and licensee disagree on the corrective action for any issue, that matter will be resolved in accordance with the Commission's procedures for backfitting requirements.

Following resolution of any comments on the draft report, the NRC Staff will request that the licensee establish and justify implementation schedules for each of the corrective actions and any ongoing analyses that may be necessary to establish appropriate corrective actions. The NRC Staff will judge the adequacy of the proposed implementation schedules based on the technical evaluation of the issues presented in the draft report and issue an implementation plan in a final report.

LICENSING ACTION AND SCHEDULE UPDATES

The final report will serve as the basis and documentation for a license amendment incorporating and formalizing the implementation schedules. The license amendment will also establish procedures to periodically update the implementation schedule.

Any new implementation requirements that arise during or following an ISAP review will be deferred, except for those issues for which the NRC Staff determines that prompt action is required to ensure the health and safety or common defense and security of the public.

The deferred implementation requirements will be evaluated collectively as part of an implementation schedule update. The update evaluations would be conducted periodically, but not more than five year intervals. The evaluation would follow the same general course as an ISAP review and would consider a revised PSA, which has been updated to reflect corrective actions and plant improvements as they are completed. The revised implementation schedule will similarly be incorporated and formalized by a new license amendment.

LIST OF SUBJECTS IN 10 CFR PART 50

Nuclear power plants and reactors, Reactor licensing criteria and reporting requirements, General Design Criteria, Backfitting, and Reactor license conditions.

The authority citation for this document is:

Dated at Washington, D.C., this day of 1984.

FOR THE NUCLEAR REGULATORY COMMISSION

Secretary to the Commission

ENCLOSURE 4

SEP PHASE II
SAFETY LESSONS LEARNED

INTRODUCTION

Phase II of the Systematic Evaluation Program consisted of detailed reviews of 137 topics for ten of the oldest domestic operating power reactors. Those reviews consisted of a comparison of the existing plant design against current licensing criteria for those topics. Subsequently, the differences from current licensing criteria were evaluated in an integrated assessment to determine their relative safety significance on a plant-specific basis.

Thus far, eight integrated plant safety assessments have been completed. As a result, a number of general lessons have been learned related to the safety of operating nuclear power reactors. For approximately two-thirds of the 137 topics reviewed, the Phase II plants either met or were found to be equivalent to current criteria. The remaining one-third of the topics usually consisted of about 70 to 90 separate safety issues. During integrated assessments, the staff concluded that no action was necessary for about one-third to one-half of the individual issues. For the remaining issues, the licensees have proposed procedural and hardware modifications, or ongoing evaluations to define the optimum corrective action, in order to improve plant safety with respect to the deficiencies identified.

The deficiencies for which corrective actions have been identified are described in three broad areas: external events, internal events and plant design. These areas are discussed below with respect to the general safety lessons learned from SEP Phase II. The staff expects that this experience is generally applicable to other operating plants which received their construction permit in the late 1960's and early 1970's.

Consequently, the staff has developed a detailed "topic list" from this experience which could be used in a follow-on program to SEP and which identifies the specific experience gained from the SEP Phase II reviews. Attachment 1 to this enclosure presents the topic list. This material supplements the general safety lessons learned and is intended to be responsive to the provisions for a summary of findings from SEP contained in the proposed amendment to the NRC Authorization Bill (H.R. 2510) from the Committee on Interior and Insular Affairs (98-103) dated May 11, 1983.

EXTERNAL EVENTS

The SEP reviews found that the existing plant designs and procedures were generally deficient in their ability to cope with local natural or man-made phenomena.

Plant Foundations: The design of older operating plants did not usually account for potential soil-structure interactions. Instances were discovered where piping runs were routed through areas with inadequate load bearing capability and building support piles were not founded in bedrock. The potential existed for soil failures to adversely affect the ability to safely shutdown the plant. These deficiencies are generally being addressed as part of larger structural upgrade programs.

Seismic Design: The SEP review found that the seismic design spectra for older plants may not have had sufficient margin. SEP developed site-specific seismic spectra based on a "uniform hazards methodology" for the evaluation of the adequacy of the seismic design of each plant. Early in the review, an issue was identified concerning anchorage of safety-related electrical equipment which was significant enough to warrant immediate action. In addition, there was generally a lack of documentation to establish the seismic capability of all electrical and mechanical equipment. Subsequent analyses were or are being performed to ensure the plants' ability to safely shutdown following a seismic event. The results thus far indicate that (1) most safety-related structures possess adequate seismic capacity, with the exception of a few structural elements (e.g., beam connections) which require strengthening; (2) most mechanical equipment is capable of withstanding postulated seismic loads, except for long-shaft vertical pumps with cast iron casings and motor-operated valves on small diameter (4" and less) pipes; (3) piping systems had previously been addressed by the Office of Inspection and Enforcement Bulletin 79-14, except for pipe supports designed by chart methods which required reanalysis for wall flexibility and anchorage to determine needed modifications.

Hydrology: The review of site-specific hydrologic characteristics found that the original plant design did not include sufficient margin to accommodate the uncertainties in severe weather conditions and related flooding and drought events. Some facilities required structural and equipment modifications to provide safe shutdown capability for flooding events. The staff also evaluated the plant emergency procedures to verify the ability to cope with various flooding levels. In most cases, the staff found the existing procedures did not assure the ability to provide for safe shutdown of the facility. The licensees have agreed to upgrade the existing flood emergency plans to incorporate appropriate actions and portable emergency equipment.

Weather-induced Loads: The review regarding the ability of the plant to withstand severe weather phenomena (i.e., precipitation, winds, and snow) concluded that each of the plants had equipment and structures which are incapable of withstanding the loadings imparted by these phenomena. The staff SEP review provided a set of site-specific wind, tornado and snow loads for each facility. Some facilities will modify the roof parapets of safety-related structures to preclude water ponding and roof failure, as a result of precipitation. The weather-induced loads are generally being addressed as part of larger structural upgrade programs.

Tornado Missiles: All of the Phase II plants were found to have external equipment or components, either safety-related or important to safety, which was not missile protected. At most plants, there was no completely protected method for safe shutdown. In each case the licensee have committed to protect at least one train of shutdown equipment, usually by means of a combination of procedural and hardware modifications.

INTERNAL EVENTS

These events deal with the ability of the plant systems and components to prevent and/or mitigate the consequences of accidents or transients. The plant systems review included the design and operation of all plant safety systems and both the electrical and mechanical aspects of shutdown, service water, and ventilation.

Safe Shutdown Capability: A review of the capability of the plant to achieve and maintain a safe shutdown condition from both inside and outside the control room using safety systems was performed. All of the Phase II plants were found to have sufficient systems available to achieve this goal; however, these systems did not always meet all of the requirements for safety systems. Revision of procedures to reach cold shutdown using both safety-grade and non-safety systems and to achieve cold shutdown from outside the control room were required. Some hardware modifications were necessary to support the procedure changes.

Service and Cooling Water Systems: The staff review of service and cooling water systems found several differences from current review criteria, particularly for the PWRs. The principal issues of concern related to single failures causing a loss of both service water pumps and to internal flooding of safety-related equipment. Several plants required further evaluation to define procedural improvements, and some hardware modifications will be necessary to enhance the reliability of service water cooling and to support the procedure changes.

Ventilation Systems: Many differences were identified with regard to the design and operation of station ventilation systems. The principal safety issue is potential common mode equipment failures due to loss of ventilation and other issues involving spread of contamination. Most of the licensees are evaluating or testing for the consequences of the loss of specific ventilation systems which are susceptible to single failures. These evaluations will be used to determine whether safety-related equipment or equipment important to safety can be expected to function following a loss of ventilation and, if not, what the consequences of equipment failure would be in order to determine the optimum corrective action. Procedural upgrades have been developed. Hardware modifications have resulted from these evaluations and tests.

System Isolation: The staff identified instances where the isolation capability between high and low pressure systems did not meet current requirements. Several plants have incorporated technical specification changes to enable pressure relief systems and at least one plant will install pressure interlocks to prevent overpressurization of the low pressure system.

Pipe Breaks: The review of pipe breaks both inside and outside containment and their effects on systems and components proved to be a major undertaking during the Phase II reviews. Most of the Phase II plants were not designed to accommodate the effects of pipe breaks inside containment. Some plants

have implemented modifications, while others are continuing to evaluate their as-built design to determine the extent of the pipe break effects and to develop corrective actions where necessary. It is expected that additional modifications will be necessary before final resolution of this issue.

Similarly, most of the Phase II plants did not comply with all of the requirements for reactor coolant pressure boundary leakage detection. The staff has concluded that reactor coolant system leakage should be evaluated in the context of the potential for and consequences of pipe breaks inside containment. In this manner, the staff feels that system sensitivity and methods for detecting leaks can be established and integrated to resolve several issues related to pipe break effects.

Accident Consequences: The staff review of potential offsite dose consequences of various postulated accidents raised a concern regarding the adequacy of existing reactor coolant system activity limits. It was determined that, for many plants reviewed, the radioiodine activity limits were much greater than those currently recommended by the Standard Technical Specifications (STS) and the activity levels actually experienced during plant operation. In order to minimize the potential consequences of accidents involving the release of primary coolant and to provide a more meaningful limit on primary coolant activity, the staff has recommended that licensees incorporate the STS limits in the plant Technical Specifications.

A review of loss-of-coolant accidents has revealed a concern with regard to the contribution to offsite dose consequences from leakage through Main Steam Isolation Valves (MSIVs) in boiling water reactors (BWRs). Although permissible leakage in the Technical Specifications has been established generically for MSIVs, excessive leakage during testing has been a common problem. The affected licensees will implement an augmented maintenance program depending on the results of a review of maintenance practices at other facilities.

Plant Chemistry: The staff review of organic materials and coatings inside containment found that most plants had some materials which could potentially degrade and exacerbate post-accident coolant chemistry. In most cases, only improved surveillance was necessary. One instance was found where the containment liner was insulated with a polyvinyl chloride (PVC) material; decomposition of PVC under accident conditions results in a release of HCl, contributing to the accumulation of hydrogen in the containment. Similar conditions may also affect the pH of the recirculated sump water, increasing the potential for intergranular stress corrosion cracking (IGSCC).

PLANT DESIGN

One of the more significant issues covered during the Phase II reviews concerned the adequacy of the overall plant design in consideration of changes in codes and standards from those in use at the time of construction and the affects of the accidents and external events described in the preceeding sections (e.g., load combinations on structures, systems and components). In addition, the staff reviewed the capability of safety systems to perform their intended functions.

Structures: The staff's review identified a number of potentially significant code changes relative to the design of structures. However, the principal issues related to the adequacy of the structures to meet the loading requirements for external phenomena. The areas of concern include hydrologic, seismic, wind and tornado, snow loadings, and load combinations. One licensee has undertaken a complete structural upgrade program designed to evaluate and upgrade the load capability of all safety-related structures. This effort has proven the benefit of an integrated plant review whereby many issues will be resolved under one program. Other licensees have implemented limited structural upgrade programs, while others are continuing to perform analyses of their structures to determine their capability and the need for modifications.

Systems and Components: In most cases the review of the quality requirements of systems and components could not be completed because of a lack of information regarding the materials or design. The principal concern related to material fracture toughness, radiographic examination, and standards for tanks, valves and pumps. These issues will be resolved by ongoing evaluations of the margins of safety in the design of equipment necessary for safe shutdown or accident mitigation and by performing the necessary volumetric examinations. Modifications have been required for some tanks and pump supports.

Containment Isolation: An extensive review of the containment isolation system was done for all SEP Phase II plants. Many instances were found where the administrative controls for isolation valves were inadequate and several valves were not locked closed (i.e., valves without mechanical locking devices). The adequacy of leakage detection capability to remotely isolate some safety systems was also identified as a potential problem.

Some branch lines were found that did not have adequate isolation. All of the licensees have agreed to implement some plant changes ranging from hardware modifications to changes in plant procedures.

Instrumentation and Control: Several issues were raised regarding the adequacy of the Reactor Protection System (RPS) isolation devices. Most plants are making hardware modifications or are performing further analyses or testing to assure proper electrical isolation. In all cases, the problem arose with the potential for faults, in systems which interface with the RPS, being propogated to the RPS and affecting its safety functions. The staff has identified such a potential in control room process recorders and indicating instruments, plant process recorders and RPS channel power supplies.

The staff also reviewed the testing requirements for the RPS system. A few cases were identified where the testing procedures were inadequate. Most of these problems related to the frequency of testing.

Power Systems: The review of plant electrical system design for most plants were found to have inadequate indication of the dc system in the control room. Several licensees have proposed modifications which improve dc system availability. Three of the licensees have agreed to modify the existing procedures regarding station battery capacity test requirements. Most of the licensees agreed to modify the diesel generator systems (e.g., bypassing diesel generator protective trips during accident conditions and providing additional annunciators and alarms). Several other concerns arose including the adequacy of protective relaying, the use of automatic bus transfers, and the administrative control of breakers during normal plant operation.

Shared Systems: Although applicable to only one of the Phase II plants, the issue of sharing safety systems between units at the same site was identified as a potential problem area. Of particular concern was the sharing of electrical systems (emergency diesel generator and station batteries). Dresden 1 and 3 share three diesels and two batteries and rely upon manual actions to assure availability of AC and DC power for postulated single failures. It was found that certain plant procedures were being used which could create a problem by transferring of faults between units or causing unavailability of safety equipment.

ENCLOSURE 4

ATTACHMENT 1

RESULTS OF SEP

Phase I of the Systematic Evaluation Program (SEP) screened approximately 800 potential safety issues for operating reactors. Many of those issues were research-oriented or components of a larger issue, such that 137 review areas or "topics" were identified for detailed review of eleven of the oldest operating reactors in Phase II (Dresden Unit 1 was subsequently deferred since the plant is indefinitely shutdown). The Phase II plants were found to meet or be equivalent to current licensing criteria for about two-thirds of the topics.

For the remaining topics, the Phase II plants were found to require procedural and hardware changes to resolve specific issues related to plant safety. In some cases, ongoing evaluations will be used to define the optimum corrective action. Moreover, many of the safety issues were found to be interrelated. All of the issues were considered collectively in an integrated plant safety assessment to ensure an effective and efficient evaluation.

The following topics were found to be significant enough in Phase II to warrant further consideration on other operating plants. The "issue" section describes the general review area and the "Safety Significance" section summarizes the specific concerns identified in the Phase II reviews and the specific corrective actions proposed. Specific sections of the Integrated Plant Safety Assessment Report (IPSAR) are referenced for the corrective actions for each plant as follows:

Palisades	NUREG-0820
Ginna	NUREG-0821
Oyster Creek	NUREG-0822
Dresden 2	NUREG-0823
Millstone 1	NUREG-0824
Yankee	NUREG-0825
Haddam Neck	NUREG-0826
LaCrosse	NUREG-0827

1.1 SETTLEMENT OF FOUNDATIONS AND BURIED EQUIPMENT

ISSUE

The objective of this review is to assure that safety-related structures, systems and components are adequately protected against excessive settlement. The scope includes the review of subsurface materials (soils or geologic) and foundations to assess the potential static and seismically induced settlement of all safety related structures and buried equipment.

SAFETY SIGNIFICANCE

Excessive settlement or collapse of foundations and buried equipment for structures, systems and components under either static or seismic loading could result in failure of structures; interconnecting piping, control systems or cables; or other equipment (tanks, etc.) such that the capability to safely shutdown the plant or mitigate the consequences of an accident could be compromised.

Millstone 1 (Section 4.2)* is evaluating the capability of piles supporting the turbine and gas-turbine buildings and the soil in areas where safety-related pipes are underlain by peat. These issues are being addressed as part of the overall evaluation of loads and load combinations. Where sufficient capacity cannot be demonstrated, the licensee will identify any necessary modifications such that safe shutdown can be assured. Similarly, support modifications are being evaluated as part of the seismic capability of San Onofre 1 because of loosely compacted backfil material (SER dated 12/1/82).

The potential for soil liquefaction at the LaCrosse site (SER dated 7/29/82) compromised the capability for decay heat removal and safe shutdown which resulted in a "Show-Cause" Order. Subsequently, the licensee installed an Emergency Service Water System and implemented associated procedures to provide safe shutdown capability.

1.2 DAM INTEGRITY

ISSUE

Dam integrity concerns the ability of a dam to prevent site flooding and ensure a cooling water supply. The safety functions would normally include remaining stable under all conditions of reservoir operation, controlling seepage to prevent excessive uplifting water pressures or erosion of soil materials and providing sufficient freeboard and outlet capacity to prevent overtopping. The objective of this topic is to assure that adequate margins of safety are available under all loading conditions and uncontrolled releases of retained water are prevented.

SAFETY SIGNIFICANCE

The safety significance of this topic is to provide the basis for assuring that all safety related structures, systems and components are adequately protected against flooding resulting from potential dam failures. Further,

*Section number refers to the section of the Integrated Plant Safety Assessment Report for the referenced plant.

the review determines whether an appropriate supply of cooling water exists in the ultimate heat sink during normal and emergency plan operations. As a result of the Phase II plant reviews, Yankee (Section 4.2) and Haddam Neck (Section 4.1.2) were identified as having potential onsite flooding due to the failure of dams. In addition, a potential loss of the ultimate heat sink has been identified for failure of downstream dams at the LaCrosse site (Section 4.4). Procedural and hardware modifications are being developed as part of ongoing evaluations to ensure safe shutdown capability.

1.3 SITE HYDROLOGIC CHARACTERISTICS AND CAPABILITY TO WITHSTAND FLOODING

ISSUE

Hydrologic considerations are the interface of the plant with the hydrosphere, the identification of hydrologic casual mechanisms that may require special plant design or operating limitations with regard to floods and water supply requirements.

The scope of this topic includes identifying the site hydrologic characteristics, the capability of structures important to safety to withstand flooding, the determination of the adequacy of the cooling water supply and the inservice inspection of water control structures. Specific issues reviewed in this topic are:

A. Hydrologic Description - To assure that plant design reflects appropriate hydrologic conditions.

B. Flooding Potential and Protection - To assure that the plant is adequately protected against floods.

C. Ultimate Heat Sink - To assure an appropriate supply of cooling water during normal and emergency shutdowns.

D. Inservice Inspection of Water Control Structures - Assure adequate inspection program is in place to prevent water control structure deterioration or failure, which could result in flooding or loss of ultimate heat sink.

SAFETY SIGNIFICANCE

Potentially serious natural flooding events, depending on the site hydrological characteristics, can cause a loss of systems and structures such that safe shutdown capability can be compromised.

In Phase II, most of the plants required some form of corrective action with respect to site flooding events. Ginna (Section 4.3 and 4.4) and Haddam Neck (Section 4.1) are providing physical protection and procedural upgrades to assure safe shutdown capability for flooding events. Oyster Creek (Section 4.1), Dresden 2 (Section 4.1) and LaCrosse (Section 4.2) will upgrade flooding emergency procedures and install scuppers on building parapets to prevent roof ponding. Oyster Creek will also install an automatic water level gauge in the intake canal.

Other plants are evaluating procedural and hardware modifications with respect to ongoing evaluations of site-specific flooding events: Millstone 1 (Section 4.1) for hurricanes and Yankee (Sections 4.1 and 4.6) for precipitation.

All plants will formalize/update their inspection programs for water control structures.

1.4 INDUSTRIAL HAZARDS

ISSUE

The objective of this topic is to ensure that the integrity of the safety related structures, systems and components would not be jeopardized due to the potential for hazards originating at nearby facilities. Such hazards include shock waves from a nearby explosion, transport of explosive gases or chemicals resulting in fires or explosions, aircraft impact and missiles resulting from nearby explosions.

Examples of activities which could result in these hazards are industrial, transportation via nearby trucking routes or shipping lanes, local, federal and military aviation and military training activities.

SAFETY SIGNIFICANCE

Industrial hazards in the vicinity of the plant site could affect safety-related structures, systems and components necessary to achieve a safe shutdown condition or to mitigate the consequences of a related accident. Although no Phase II plants have been required to implement modifications, many cases were identified where new airports, air traffic patterns or transportation routes could have affected plant safety. In addition, other industrial growth could potentially affect plant operation (e.g., LNG plant near Calvert Cliffs).

1.5 TORNADO MISSILES

ISSUE

Plants designed after 1972 have been designed for protection against tornadoes. The concern exists, however, that plants constructed prior to 1972 may not be adequately protected, in particular those reviewed before 1968 when AEC criteria on tornado protection was first developed.

The scope of this topic is to assure that safety structures, systems and components can withstand the impact of an appropriate postulated spectrum of tornado generated missiles.

An assessment of the ability of a plant to withstand the impact of tornado missiles would include:

1. Determination of the capability of the exposed systems, components and structures to withstand key missiles (including small missiles with penetrating characteristics and larger missiles which result in an overall structural impact), and

2. Determination of whether any areas of the plant require additional protection.

SAFETY SIGNIFICANCE

The failure of safety related structures, systems or components due to a tornado-induced missile could compromise the ability of the plant to safely shutdown. In SEP Phase II, a number of systems necessary for safe shutdown were identified in several plants which had no protection from tornado missiles. All Phase II plants required some form of corrective action to protect against tornado missiles to ensure that the plant could be safely shutdown. Millstone (Section 4.7), Haddam Neck (Section 4.6), Yankee (Section 4.8), Ginna (Section 4.11.2), and LaCrosse (Section 4.9) are evaluating procedures and alternatives. It is the staff's position that the licensees should demonstrate the ability to achieve and maintain a safe shutdown condition using at least one train systems and/or components protected from tornado missiles. Oyster Creek (Section 4.6) has committed to implement hardware modifications to assure a safe shutdown path. Big Rock Point and San Onofre have a substantial number of unprotected systems and the staff expects resolution to involve hardware modifications. Dresden 2 (Section 4.5) is currently evaluating systems and procedures and it is expected that only procedural changes will be necessary.

1.6 TURBINE MISSILES

ISSUE

A number of non-nuclear plants and two nuclear plants have experienced turbine disc failures. Also, one plant has had chemistry problems leading to sodium deposits which caused stress-corrosion cracking of turbine discs. Failure of turbine discs and rotors can result in high energy missiles which have the potential for causing failures in safety related systems.

Two areas of concern should be considered:

A. Design overspeed failures - material quality of disc and rotor, inservice inspection for flaws, chemistry conditions leading to stress corrosion cracking, and

B. Destructive overspeed failures - reliability of electrical overspeed protection system, reliability and testing program for stop and control valves, inservice inspection of valves.

The focus on the review is on turbine disc integrity and overspeed protection, including stop, intercept and control valve reliability.

SAFETY SIGNIFICANCE

Turbine generated missiles may strike safety related systems and components and jeopardize the ability of the plant to safely shutdown. Cracks have been discovered in turbines which could contribute to the possibility of generating turbine missiles. SEP Phase II reviews resulted in Oyster Creek (Section 4.7), Millstone 1 (Section 4.8) and Dresden 2 (Section 4.6) upgrading the inspection program for turbine discs and overspeed protection systems to preclude the potential for missile generation.

2.1 SEVERE WEATHER EFFECTS ON STRUCTURES

ISSUE

Safety related structures, systems and components should be designed to function under all severe weather conditions to which they may be exposed. Meteorological phenomena to be considered include straight wind loads, tornadoes, snow and ice loads, and other phenomena judged to be significant for a particular site.

The objective of this topic is to identify those meteorological conditions which should be considered in the SEP structural reviews to determine the ability of structures to withstand conditions, such as, flooding, wind, tornadoes, hurricanes, tsunamis, and seiches. The review includes the dynamic effects of waves, tornado pressure drop loading and possible in-leakage due to floods.

SAFETY SIGNIFICANCE

During SEP Phase II, site specific evaluations of windspeed versus recurrence interval, and snow loads were developed to assess the appropriate loading conditions to be assumed in the structural evaluations described in Section 2.2. This evaluation is intended to ensure that structures, systems and components are adequately designed to resist the loads imposed by severe weather phenomena. In many instances in SEP Phase II, the potential flood, wind and tornado loads have been found to be greater in magnitude than those considered in the original design, resulting in the need for plant modifications. Ginna (Section 4.2) was required to implement a groundwater level monitoring program. Oyster Creek, Millstone 1 and Yankee, depending on the results of continuing evaluations, will have to provide protection against wind loads, hydrostatic forces, floating potential or flood levels. In addition, the effects of severe weather are being considered in both the Ginna and Haddam Neck structural upgrade programs. The staff expects that San Onofre and Big Rock Point will require similar corrective actions.

2.2 DESIGN CODES, CRITERIA AND LOAD COMBINATIONS FOR STRUCTURES

ISSUE

Structures important to safety should be designed, fabricated, erected and tested to quality standards commensurate with the safety function to be performed. Also, structures that are required to withstand the effects of a safe shutdown earthquake and remain functional should be classified as Seismic Category I. Due to the evolutionary nature of design codes and standards, operating plants may have been designed to codes and criteria which differ from those currently used for evaluating new plants.

The objective of this topic is to determine the classification of structures applicable at the construction permit stage and at present and to provide assurance that plant Category I structures will withstand the appropriate design conditions without impairment of structural integrity or the performance of required safety functions. The design codes, criteria and load combinations for all Category I structures (i.e., containment, structures inside containment and structures outside containment) are reviewed.

SAFETY SIGNIFICANCE

Due to the evolutionary nature of structural codes and standards, it is possible that some aspects of the original plant design are less conservative than those required by current codes.

In Phase II, detailed comparisons of earlier versions to current American Concrete Institute and American Institute of Steel Construction and American Society of Mechanical Engineers (ASME) standards identified a number of potentially significant changes which resulted in the need for some plant modifications.

In addition, new or higher loading conditions have been identified for seismic, winds, rainfall, snow and other severe external events. The effects of these loading conditions on load combinations has been evaluated in conjunction with that for the code changes described above for: Palisades (Section 4.12), Ginna (Section 4.17), Oyster Creek (Section 4.12), Dresden 2 (Section 4.10), and LaCrosse (Section 4.14). In some cases the effects of load combinations are being considered in the application of a overall plant structural upgrade program.

2.3 CONTAINMENT DESIGN AND INSPECTION

ISSUE

The purpose of this topic is to review the inspection program of prestressed concrete containments. The program should include liftoff testing and acceptance criteria, testing or prestressing tendons and possible deterioration of prestressed containments. The likelihood of delamination occurring in the shell walls or dome is also reviewed.

SAFETY SIGNIFICANCE

Should the tendons of prestressed concrete containment vessel experience faster than normal relaxation or corrosion, or should the concrete degrade because of delamination, the ability of the containment to withstand the design loads may be compromised. The two plants for which this topic was applicable in SEP Phase II required improved inspection programs to detect and correct such degradation: Palisades (Section 4.11 and 4.13) and Ginna (Section 4.16).

2.4 SEISMIC DESIGN OF STRUCTURES, SYSTEMS AND COMPONENTS

ISSUE

The objective of this topic is to review and evaluate the original seismic design (seismic input, analysis methods, design criteria, seismic instrumentation, seismic classification) of the safety-related plant structures, systems and components to ensure the capability of the plant to withstand the effects of an earthquake. Further, the topic will assure that the free field ground motion specified for the plant design adequately represents the vibratory ground motion associated with a postulated safe shutdown earthquake (SSE) at the plant. The free field ground motion will be represented as the design spectra for the facility that will be utilized as the input to analyses to verify the design adequacy of structures, piping and equipment. This review and evaluation will address the SSE only, since it represents the most severe event that must be considered in the plant design. The scope of the review includes three major areas: the integrity of the reactor coolant pressure boundary; the integrity of fluid and electrical distribution systems related to safe shutdown; and the integrity of mechanical and electrical equipment and engineered safety features systems (including containment). A detailed review of all safety related structures, systems and components will not be conducted; rather a sampling approach supported by a set of confirmatory analyses will be performed. The sample size and confirmatory analyses will be increased, if necessary.

SAFETY SIGNIFICANCE

Many nuclear power plant facilities received construction permits in the 1960's. Seismic design criteria and procedures evolved significantly during and after this period.

In order to ensure that a facility can withstand a large earthquake without the loss of capability of systems important to safety and the structures that house them, it is necessary to verify that the free field ground motion has been properly defined. The effects from such external phenomena on a nuclear plant are particularly significant in that the entire facility is affected. In SEP Phase II, the free field seismic ground motion was found to be conservatively specified by the licensees in some cases but not in others. As a result, the NRC identified the appropriate conservative spectra and historic seismicity for each plant site. Results of seismic analyses using the free field ground motion defined as a part of SEP Phase II have identified the need for hardware modifications to some plant systems and structures.

As a result of the SEP Phase II seismic review, four common design deficiencies of plant facilities were identified: (1) structural integrity of anchorage and support systems of all safety related electrical equipment, (2) structural integrity of vertical pumps, (3) field erected tanks, and (4) support systems of small diameter safety related piping with large valve operators.

In most cases, improved seismic resistance modifications were made during the Phase II topic review and ongoing evaluation programs were defined to determine additional plant modifications required to ensure safe shutdown capability following a design-basis seismic event: Palisades (Sections 3.3.1 and 4.10), Ginna (Sections 3.32 and 4.15), Oyster Creek (Section 4.11), Dresden 2 (Section 4.9), Millstone 1 (Section 4.11), Yankee (Section 4.11), Haddam Neck (Section 4.10), and LaCrosse (Section 4.13).

In addition, because of issues raised regarding the seismic design basis for San Onofre 1, the licensee has initiated a major seismic upgrade program and a confirmatory order was issued to maintain the plant in a shutdown condition until specified modifications are completed.

3.1.1 SHUTDOWN SYSTEMS

ISSUE

The safety objective of this topic is to ensure reliable plant shutdown capability using safety-grade equipment. Systems and components important to safety should be designed, fabricated, erected and tested to quality standards commensurate with the safety function to be performed. Also, systems and components that are required to withstand the effects of a safe shutdown earthquake and remain functional should be classified as Seismic Category I. Due to the evolutionary nature of design codes and standards, operating plants may have been designed to requirements that are not as conservative as those currently required. Systems needed to remove decay heat and reach safe shutdown should have sufficient redundancy to assure that the system function can be accomplished with a loss of offsite power and a single failure. Systems needed to shutdown must also remain functional following external events. This topic identifies the necessary systems and determines the classifications applicable at the construction permit stage and at present. In addition, the plant operating procedures which direct the use of these systems during normal and abnormal events will be evaluated.

SAFETY SIGNIFICANCE

Due to the evolutionary nature of component and system codes, it is possible that some aspects of the original plant design are sufficiently less conservative than that required by current codes.

In Phase II, it was impossible to document the seismic and quality group classification of all safety related components/systems at all the plants. For the affected components/systems, additional analyses were required to justify continued use or replacement.

A reliable method for decay heat removal is needed to ensure capability to safely shutdown the reactor. The plant must have the capability to safely shutdown following external events. This topic identifies the required functions and systems.

It was the experience of Phase II for all plants that some of the systems normally used for safe shutdown were not adequately protected for external events. The safety significance of these events is discussed in the appropriate topic definitions relating to external events. Some systems were seismically qualified, but not protected for floods or wind and tornado effects. Thus, no single shutdown method or set of systems met all of the review criteria. The SEP review evaluated operation of alternative systems that could be used in the event of failures, however, the operating procedures had to be revised to address use of these systems: Palisades (Sections 4.16.2 and 4.24.1), Ginna (Section 4.21.2), Oyster Creek (Section 4.18), Dresden 2 (Section 4.25.2), Millstone (Section 4.17), Haddam Neck (Sections 4.19.2 and 4.19.4), and LaCrosse (Sections 4.19.1 and 4.19.3). Several SEP Phase II licensees have commented that this review was particularly useful and worthwhile.

3.1.2 SHUTDOWN ELECTRICAL INSTRUMENTATION AND CONTROLS

ISSUE

The electrical, instrumentation and control features of systems required for safe shutdown, including support systems, are reviewed to determine whether they meet current licensing requirements. The review also includes the capability and methods of bringing the plant from a high pressure condition to a low pressure cooling condition assuming the use of only safety equipment.

SAFETY SIGNIFICANCE

A reliable method of decay heat removal is necessary to ensure the capability to safely shutdown the reactor for all design basis conditions. This topic identifies the instrumentation and controls necessary to accomplish this function and evaluates their quality with respect to IEEE Standard 279-1971.

It was the experience of the SEP Phase II review that most plants failed to satisfy the acceptance criteria and that system and technical specification modifications were required. All plants were required to provide at least one protected path to achieve safe shutdown. (See Section 3.1.1 for related Phase II experience.) This protected path includes the required instrumentation.

In addition, Palisades (Section 4.24.2) and Ginna (Section 4.25.2) were required to install a redundant sensor for component cooling water surge tank level, LaCrosse (Section 4.19.2) will install a second level controller for the shutdown condenser. Palisades (Section 4.24.1) is developing procedures for stripping non-essential loads from a recently-installed higher capacity battery to ensure instrumentation will be powered until the delayed access offsite power line can be made available.

3.2 SERVICE AND COOLING WATER SYSTEMS

ISSUE

The objective of this topic is to assure that the station service and cooling water systems have the capability, with adequate margin, to meet their design objective. To assure, in particular, that:

A. Cooling water systems are capable of transferring heat from structures, systems and components important to safety to the ultimate heat sink.

B. Systems are provided with adequate physical separation such that there are no adverse interactions among those systems under any mode of operation.

C. Sufficient cooling water inventory has been provided or that adequate provisions for makeup are available.

SAFETY SIGNIFICANCE

The service and cooling water systems remove decay heat from the core under normal and emergency conditions to the ultimate heat sink. Loss of these systems could result in a core melt event. In SEP Phase II, deviations from the acceptance criteria were found that resulted in modifications to plants and plant procedures. Palisades (Section 4.27) was required to show that adequate water flow is possible given diesel generator failure. Ginna (Section 4.25) was required to install a level sensor in the component cooling water tank and modify the technical specifications for the two service water system pumps in operation so that they will not be serviced by the same diesel generator. Haddam Neck (Section 4.32.4) was required to show that adequate procedures and time exist to balance service water flow if non-critical loads are not isolated. San Onofre may require hardware modifications and/or procedural or technical specification changes regarding the salt water cooling water system.

3.3 VENTILATION SYSTEMS

ISSUE

To assure that the ventilation systems have the capability to provide a safe environment for plant personnel and for engineered safety feature systems, it is necessary to review the design operation of these systems. For example, the function of the spent fuel pool area ventilation system is to provide ventilation in the spent fuel pool equipment areas, to permit personnel access, and to control airborne radioactivity in the area during normal operation, anticipated operational transients, and following postulated fuel handling accidents. The function of the engineered safety feature ventilation systems is to provide a suitable and controlled environment for engineered safety feature components following certain anticipated transients and design basis accidents.

SAFETY SIGNIFICANCE

The safety significance of a ventilation system is based on the safety importance of the systems and/or components for which ventilation is needed. In SEP Phase II, deviations from the acceptance criteria were found resulting in the need for hardware modifications and evaluation of procedural analyses. Palisades (Section 4.28) had to demonstrate the operability of the auxiliary feed pump ventilation and demonstrate the effects of loss of ventilation and determine whether certain safety related systems can operate upon loss of ventilation. Yankee (Section 4.31) was required to modify the diesel generator building vent to eliminate the potential for single failure. Millstone (Section 4.32) was required to evaluate the effects of loss of ventilation on safety-related systems to determine procedural changes. Big Rock Point (Section 4.26) was required to evaluate pocketing of hydrogen in the electrical equipment room following a loss of offsite power (maximum battery offgassing on charge from diesels and a coincident loss of ventilation). Big Rock Point has also discovered that the natural circulation cooling of the diesel generator space was not sufficient for long term power operations. Procedural and/or hardware modifications are also expected for San Onofre 1 for rooms containing switchgear, batteries and other equipment.

3.4 ISOLATION OF HIGH AND LOW PRESSURE SYSTEMS

ISSUE

Several systems that have a relatively low design pressure are connected to the reactor coolant pressure boundary. The valves that form the interface betwen the high and low pressure systems must have sufficient redundancy and interlocks to assure that the low pressure systems are not subjected to pressures that exceed design limits for these systems. This problem is complicated, because under certain operating modes (e.g., shutdown cooling and ECCS injection), these valves must open to assure adequate core cooling capability.

As a specific example, a number of plants have residual heat removal (RHR) systems in which the design pressure rating is lower than that of reactor coolant system (RCS) pressure boundary to which the system is connected. The RHR system is normally locked outside of primary containment and has motor-operated valves (MOVs) which isolate it from the RCS. Therefore, there is a potential that these systems will be subjected to pressure stresses in excess of their design rating should the isolation MOVs be opened inadvertently while the RCS is above the RHR system design pressure rating. For some plants this could result in a loss-of-coolant accident outside containment. Generally, interlocks are provided to prevent opening these isolation MOVs under high RCS pressure conditions. Some plants may not have appropriate interlocks to prevent opening, or to provide automatic closure if the pressure increases while the isolation MOVs are open, or other features to assure that failure of the low pressure system boundary is unlikely.

SAFETY SIGNIFICANCE

The failure of systems that isolate low pressure systems from high pressure systems can lead to an uncontrolled loss of coolant accident outside containment. It was the experience from SEP Phase II reviews that most of the RHR systems and many of other interface systems failed to satisfy the acceptance criteria and that systems and procedure modifications were required to assure adequate overpressure relief capability for the low pressure system. For PWRs adequate RHR relief capacity did not exist for postulated pressure transients starting from low pressure. Palisades (Section 4.16.1), Ginna (Section 4.21.1), Haddam Neck (Section 4.19.1), and San Onofre will modify Technical Specifications to ensure that the overpressure protection system is operable whenever the RHR system is connected to the reactor coolant system. In addition, Ginna (Section 3.3.3.1) has established a check valve test program, Oyster Creek (Section 4.19) is to provide relief/capacity for the reactor water cleanup (RWCU) system, Palisades (Section 4.17) is required to provide verification of check valve closure after each use of the low pressure safety injection system, Millstone 1 (Section 4.18) and Yankee (Section 4.20) will provide an independent interlock for low pressure systems, and Haddam Neck (Section 4.20) will install pressure sensor interlocks on both low pressure and high pressure safety injection system isolation valves.

3.5 AUTOMATIC ECCS SWITCHOVER

Most PWRs require operator action to realign ECCS systems for the recirculation mode following a LOCA.

Current guidelines state that automatic transfer to the recirculation mode is preferable to manual transfer. However, a design that provides manual initiation is sufficient provided that adequate instrumentation and information display are available for the operator to transfer at the correct time.

The review, therefore, also addresses the procedures provided for manual switchover, the instrumentation available, and possible operator errors and consequences.

SAFETY SIGNIFICANCE

The failure of systems used to change from injection to recirculation or failure of the operator to complete the switchover in a timely manner can result in the loss of emergency core cooling and/or residual heat removal following a loss of coolant accident.

As a result of the SEP Phase II reviews, Ginna (Sections 3.3.5 and 4.23.1) was required to install a redundant RWST level alarm and improve procedures for emergency safety features switchover. At Haddam Neck (Section 4.23) the licensee has committed to install a redundant level indicator on the RWST and add an alarm to alert the operators when to start the switchover from

injection to recirculation. Further, the licensee is also reviewing their procedures to assure that adequate time exists for the operator to manually accomplish the switchover procedures. San Onofre is also expected to require instrument modifications and/or procedural changes.

3.6.1 EMERGENCY AC POWER SYSTEMS

ISSUE

The electrical independence of redundant safety related onsite power sources must provide redundancy and independence of safety related power sources necessary to meet the single failure criterion.

Diesel generators, which provide emergency standby power for safe reactor shutdown in the event of total loss of offsite power, have experienced a significant number of failures. The failures to date have been attributed to a variety of causes, including failure of the air startup, fuel oil, and combustion air system.

The review includes the reliability of protection interlocks and testing of diesel generators to assure that the diesel generator system meets the availability requirements for providing emergency standby power to the engineered safety features as well as the independence of onsite power distribution systems and features such as automatic bus transfers and breaker connections that could affect independence of redundant trains.

SAFETY SIGNIFICANCE

The failure of the onsite power systems can result in the loss of core cooling systems. Electrical systems that fail to satisfy the acceptance criteria may compromise the independence of all systems that are required for safety.

In SEP Phase II, several plant electrical distribution systems and onsite generation systems failed to satisfy the acceptance criteria. Oyster Creek (Section 4.25) will evaluate the automatic bus transfer and identify the corrective actions necessary to insure that faulted loads will not be transferred; Dresden 2 (Section 4.21) is to verify the adequacy of their protective relaying and provide procedures to verify that disconnects links are open and that the breakers are not used during operation; Millstone 1 (Section 4.23) will install interlocks and evaluate the automatic bus transfer. Haddam Neck (Section 4.24) has proposed to install protective devices to prevent transferring faults between redundant power divisions. LaCrosse (Section 4.24) will install an interlock on two breakers to prevent paralleling of the two diesel generators.

Palisades (Section 3.3.3) was required to provide separate annunciators for the status of the safety systems. Oyster Creek (Section 4.31) will modify diesel generator annunciators and will evaluate installation of bypasses of selected diesel generator protective trips under accident conditions. Dresden 2 (Section 4.26) and Haddam Neck (Section 4.9) will install bypasses for the diesel generator trips as above and Millstone (Section 4.28) will provide such bypasses for the emergency generators as well as an improved preventive maintenance program for the gas turbine generator.

3.6.2 EMERGENCY DC POWER SYSTEMS

ISSUE

The electrical instrumentation of redundant safety related onsite power sources must provide redundancy and independence of safety related power sources necessary to meet the single failure criterion.

In addition, the dc power system battery, and bus voltage monitoring and annunciation design is evaluated with respect to dc power system operability status indication to the operator so that timely corrective measures can be taken in the event of a loss of an emergency dc bus.

SAFETY SIGNIFICANCE

A failure of the dc power systems can lead to station blackout (see NUREG-0666 for additional information).

In SEP Phase II, most plants were required to provide additional dc system alarms and indication in the control room: Palisades (Section 3.3.4), Ginna (Section 4.24), Oyster Creek (Section 4.32), Dresden 2 (Section 4.28), Millstone 1 (Section 4.30.1), Yankee (Section 4.28.1), Haddam Neck (Section 4.31), and LaCrosse (Section 4.28).

Technical Specifications will be revised to reduce the time a plant may operate with a battery system inoperable at Dresden 2 (Section 4.21.4) and Millstone 1 (Section 4.30.2).

For three plants, Dresden 2 (Section 4.21), Millstone 1 (Section 4.23), and Haddam Neck (Section 4.24.2), administrative procedures and/or physical devices will be used to prevent tie breaker, disconnect link or automatic bus transfer malfunctions from paralleling redundant trains of dc power.

Periodic testing of battery capacity will be provided for Palisades (Section 4.25), Ginna (Section 3.3.8), Millstone 1 (Section 4.29), and Haddam Neck (Section 4.30).

3.7 RCPB LEAKAGE DETECTION

ISSUE

The safety objective of this topic is to determine the reliability and sensitivity of leakage detection systems which monitor the reactor coolant pressure boundary. The leakage detection systems should monitor reactor coolant pressure boundary leakage to the containment and to interconnecting systems.

SAFETY SIGNIFICANCE

The safety significance of a reliable and sensitive leakage detection system is that it provides the reactor operator with an adequate margin of time to initiate procedures to identify the source of a leak, and isolate and repair it before the leak can grow to a size that might result in loss of component or system function or a loss of coolant accident. In Phase II, no plant completely met the current licensing criteria for this topic. Issues related to system sensitivity were integrated with the evaluation of the effects of pipe breaks inside containment in order to determine critical leakage rates. Revisions to procedures or Technical Specifications were required to reflect system limitations and to enhance system reliability. Such corrective actions were relatively consistent for most of the Phase II plants: Palisades (Section 4.15.2), Oyster Creek (Section 4.16), Dresden 2 (Section 4.13), Millstone 1 (Section 4.16), Yankee (Section 4.16), Haddam Neck (Section 4.16), and LaCrosse (Section 4.17.2).

3.8 SHARED SYSTEMS

ISSUE

The sharing of engineered safety feature systems (ESF), including onsite emergency power systems, and service systems for a multiple unit facility can result in a reduction of the number and of the capacity of onsite systems to below that which is needed to bring either unit to a safe shutdown condition or to mitigate the consequences of an accident. The review of these shared systems for multiple unit stations should include equipment powered from each of the units involved.

SAFETY SIGNIFICANCE

The conflicting needs created by an accident in one unit and the need to safely shutdown a second unit may result in a loss of system capability necessary to accomplish both functions simultaneously.

As a result of SEP Phase II, Dresden 2 (Section 4.23) was prohibited from paralleling shared dc systems or placing diesel generator 2/3 switch in "bypass" during power operation in order to ensure emergency power reliability for Dresden Units 2 and 3.

4.1 REACTOR PROTECTION SYSTEM AND ENGINEERED SAFETY FEATURE SYSTEMS ISOLATION

ISSUE

Non-safety systems generally receive control signals from the reactor protection system (RPS) and Engineered Safety Feature (ESF) sensor current loops. The non-safety circuits are required to be isolated to insure the independence of the RPS and ESF channels. Requirements for the design and qualification of isolation devices are quite specific. Evaluation of the quality of isolation devices is not the safety issue of concern; rather the issue to be reviewed is the existence of isolation devices which will preclude the propagation of non-safety system faults to safety systems.

SAFETY SIGNIFICANCE

The lack of isolation devices, in the event of non-safety system failures, could result in the propagation of faults to safety systems and common cause failures may result. During SEP Phase II, it was determined that RPS systems for Oyster Creek (Section 4.27), Dresden 2 (Section 4.24) and Millstone 1 (Section 4.25) had no isolation from non-safety systems (e.g., plant computer), that multiple channels of the RPS were involved, and that the system modifications were necessary.

Palisades (Section 4.23.1) was required to install qualified isolation devices on steam generators A and B pressure channels and on a reactor coolant flow channel to the plant computer. Haddam Neck (Section 4.26) and LaCrosse (Section 4.26.3) are evaluating existing isolation provisions and will install isolation devices as required. LaCrosse (Section 4.26.4) will separate power supplies to the full scram channels.

4.2 RPS AND ESF TESTING

ISSUE

The plant design must assure that all Emergency Core Cooling System (ECCS) components, including the pumps and valves, are included in the component and system test, that the frequency and scope of the periodic testing is identified, and that the test program will provide adequate assurance that the system will function when needed.

SAFETY SIGNIFICANCE

The lack of an adequate integrated test program for safety-related instrumentation and control systems may invalidate the design of the safety systems because only random failures are considered in single failure analyses. If failures exist that cannot be detected by test the system function may be defeated. A periodic test program which identifies pre-existing failures will significantly improve system reliability.

In SEP Phase II, several plants failed to satisfy the IEEE 279-1971 testing criteria resulting in the need for system and procedure modifications. In many instances, testing that is performed by procedure was added to the Technical Specifications. These cases include Oyster Creek (Section 4.23 and 4.26.2), Yankee (Section 4.24), Haddam Neck (Section 4.25), and LaCrosse (Section 4.23).

Operating procedures were modified to include channel checks on average power range monitor (ARPM) - flow based on APRM reduced high flux channels for Millstone 1 (Section 4.24.1).

The Technical Specifications for Yankee (Section 4.23) were revised to include testing of automatic valves in the flow path of the emergency core cooling system.

Oyster Creek (Section 4.23) has modified testing procedures for the emergency condenser logic train such that each train can be tested separately while at power.

5.1 ORGANIC MATERIALS

ISSUE

The design basis for the selection of paints and other organic materials is not documented for most operating reactors. The plant design must assure that organic materials, such as organic paints, coatings and insulation materials, used inside containment do not adversely affect the operation of the engineered safety feature equipment inside containment during accidents when they may be exposed to high tempertaures, steam environments, high radiation fields, and containment spray systems.

The scope of this review will include an evaluation of qualification tests and licensee inspection and repair programs to assure that the organic materials will maintain their integrity and remain in a serviceable condition after exposure to the extreme environmental conditions of a design basis accident.

SAFETY SIGNIFICANCE

The degradation of unqualified organic materials under accident conditions could contribute corrosive materials to the containment environment and debris that could impair the function of systems necessary to mitigate the consequences of an accident. During SEP Phase II, some plants used organic paints and insulation containing leachable materials inside containment which required both procedural (i.e., inspection) and plant modification. Oyster Creek (Section 4.21) was required to inspect and repair drywell coatings and to recoat the torus.

Yankee (Section 4.21.2), Big Rock Point (Section 4.19.1) and San Onofre 1 will institute a periodic coating inspection program.

Big Rock Point (Section 4.19.2) will periodically sample lake water and the standby liquid control system for chlorides and develop contingency plans for corrective actions should high levels be found.

5.2 RCS WATER PURITY (BWR)

ISSUE

The reactor water cleanup systems in direct-cycle BWR plants, in conjunction with the primary water monitoring system, must have the capability to remove contaminants introduced by main condenser leakage.

SAFETY SIGNIFICANCE

A failure to remove contaminants and maintain water purity could lead to intergranular stress corrosion cracking of austenitic stainless steels, causing an accelerated degradation of the reactor coolant pressure boundary and reactor internal components. Oyster Creek (Section 4.20) was required to implement proposed water purity procedure limits. Millstone 1 (Section 4.19.1) is to revise their technical specifications to incorporate Regulatory Guide 1.56 purity limits and to incorporate procedures for maintaining minimum reserve capacity in the reactor water cleanup system and condensate system demineralizers.

6.1 PIPE BREAK EFFECTS ON SYSTEMS AND COMPONENTS

The staff will review the pipe break and crack location criteria and methods of analysis for evaluating the dynamic effects associated with postulated breaks and cracks in high- and moderate-energy fluid system piping, including "field run" piping, inside and outside containment. Further, the effects of postulated pipe break on the integrity and function of systems and components relied upon for safe reactor shutdown to mitigate the consequences of a postulated pipe break will also be reviewed. The review includes the general layout of high energy piping systems inside containment and moderate-energy systems inside and outside containment with respect to the plant arrangement, identification of protective structures and piping restraints, and the assumptions made in the analyses of (1) the availability of offsite power, (2) a single failure, (3) other special provisions applicable to certain dual purpose systems, and (4) the use of available systems to mitigate the consequences of piping failure. The effects of postulated failures, in non-seismic low-energy fluid systems both inside and outside containment which could lead to internal flooding of essential systems and components are also reviewed.

SAFETY SIGNIFICANCE

Systems needed to achieve a safe shutdown following a postulated pipe break must be capable of withstanding the consequences of the event. In SEP Phase II, it was determined that breaks in moderate energy lines and some high-energy fluid systems outside containment were not reviewed as part of earlier NRC generic reviews for some operating reactors. Modifications have been

implemented to assure protection for internal flooding and the effects of sprays due to cracks or breaks in moderate energy lines. At some plants these events had common mode effects resulting in the potential loss of a safety function. For high-energy pipe breaks inside containment no previous review had been done. Locations have been identified where adverse interactions could occur such that ECCS operation and safe shutdown capability would be compromised.

Specification modifications made as a result of Phase II evaluations are: Ginna (Sections 3.3.1, 4.13 and 4.14) will relocate cables inside containment, replace steam heating lines with electric resistance heaters and provide jet shields for steam heating lines in the auxiliray building. Palisades (Section 4.27.2) will provide spray protection for service water pumps, Yankee (Section 4.10) will modify their ISI program to include 4 welds on non-return line valves and install a jet impingement plate on switchgear room wall. All plants are still conducting evaluations of the effects of pipe break for certain postulated break locations to determine where other plant modifications are necessary.

6.2 CONTAINMENT ISOLATION

ISSUE

Isolation provisions for lines that penetrate the primary containment maintain an essential leaklight barrier against the uncontrolled release of radioactivity to the environment and also, prevent the uncontrolled release of primary system coolant as a result of postulated pipe breaks outside containment. The isolation function must be accomplished without endangering the performance of post-accident safety systems.

SAFETY SIGNIFICANCE

The containment isolation system provides the final barrier to the release of radioactivity to the environment. The isolation provisions must be capable of performing this function for a spectrum of accident and transient events with a single failure. In SEP Phase II, significant deviations from the acceptance criteria resulted in the need for both procedural and hardware modifications.

Isolation valves will be added to selected lines at Ginna (Section 4.22.2), Dresden 2 (Section 4.18.6), Millstone 1 (Section 4.20.2), Yankee (Section 4.22.1), and LaCrosse (Sections 4.21.3 and 4.21.2.3).

Locking devices and/or administrative controls will be used to ensure manual isolation valves are locked closed at Ginna (Section 4.22.4), Oyster Creek (Section 4.22.1), Dresden 2 (Section 4.18.1), Millstone 1 (Section 4.20.1), Haddam Neck (Section 4.22.4), Big Rock Point (Section 4.20), and LaCrosse (Section 4.21.3).

In essential systems with remote manual isolation valves, provisions for leakage detection and suitable procedures will be provided at Oyster Creek (Section 4.22.2), Dresden 2 (Section 4.18.2) and Millstone 1 (Section 4.20.3). Procedures for manual isolation of some lines will be developed for LaCrosse (Section 4.21). Valve actuation was changed from manual to power - operated for one valve and a threaded pipe connection will be removed at Palisades (Section 4.20.3 and 4.20.4).

The isolation provisions of some lines are still being evaluated for several plants (Haddam Neck, Yankee, Big Rock Point, and San Onofre) to determine where additional modifications are necessary.

6.3 RCS SPECIFIC ACTIVITY LIMITS

ISSUE

The coolant activity levels have a proportionate effect on those accidents involving primary coolant release (without core damage) to the environment. Implementation of Standard Technical Specifications (STS) limits are usually adequate to alleviate the concerns regarding resultant offsite doses.

The scope of the topic will be to examine the plant technical specifications to determine the degree of compliance with the appropriate STS. An evaluation may be performed to determine the adequacy of the existing plant technical specification limits in restricting offsite dose. The review will cover those accidents whose primary dose contribution is from reactor coolant leakage to the atmosphere (e.g., main steam line break outside of containment, steam generator tube rupture and small line breaks outside containment).

A determination of the plant specific atmospheric dispersion factor may be required to evaluate the offsite dose consequences.

SAFETY SIGNIFICANCE

The primary coolant activity is a measure of the core integrity during normal plant operation. Excessive activity would suggest a degradation of the fuel and could lead to excessive offsite doses in the event of an accident involving a release of primary coolant.

Oyster Creek (Sections 4.36 and 4.37), Dresden 2 (Sections 4.31 and 4.32) and Millstone 1 (Sections 4.35 and 4.36) will revise plant Technical Specification limits on dose-equivalent iodine-131 in the primary coolant to reduce radiological consequences of line breaks outside containment.

Haddam Neck (Sections 4.35, 4.38 and 4.39) is implementing lower Technical Specification limits on primary and secondary iodine concentrations.

ENCLOSURE 5

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

July 12, 1983

Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

SUBJECT: PROPOSED INTEGRATED SAFETY ASSESSMENT PROGRAM (ISAP)

Dear Dr. Palladino:

During the 279th meeting of the ACRS, July 7-9, 1983, the NRC Staff described the Integrated Safety Assessment Program (ISAP) that is being proposed as an alternative to the extension of the Systematic Evaluation Program (SEP) and to the National Reliability Evaluation Program (NREP).

The proposed ISAP is intended to provide an integrated evaluation and resolution of pending licensing actions and safety issues for operating reactors. It will use the results from a plant-specific probabilistic safety assessment (PSA) and a comprehensive review of plant operating experience and management performance. It will include extensive participation by the licensee in the safety reviews and assessments and in the preparation of the PSA. The NRC Staff participation will involve a team effort and will emphasize knowledge and understanding of the design of each plant being reviewed.

Although the proposed program has been developed well beyond the conceptual stage, many specific aspects of implementation remain to be developed. We approve the concept and the program as it is now proposed, and we believe that acceptable and workable details for implementation can be developed. We believe that this program can lead to more rational and more effective regulation, and would enhance the protection of the health and safety of the public.

We would expect to review the details and implementation of this program and to participate otherwise in the same way that we have participated in the SEP.

Sincerely,

J. J. Ray
Chairman

ENCLOSURE 1

(7590-01)

NUCLEAR REGULATORY COMMISSION
10 CFR PART 50

Commission Policy Statement on the
Systematic Safety Evaluation of Operating
Nuclear Power Reactors

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of Commission Policy Statement.

SUMMARY: This Policy Statement describes a pilot program for which the Commission has developed the regulatory policies and practices to conduct integrated assessments for operating nuclear power reactors. This program is called the Integrated Safety Assessment Program (ISAP) and will address significant regulatory requirements which have evolved since the plant was originally licensed and pending licensing actions which have evolved from a variety of other sources. An integrated assessment will be conducted on a plant-specific basis, as part of a trial program, to evaluate all licensing issues on a given facility and to establish schedules for any necessary plant improvements. In addition, procedures have been established to allow for a periodic updating of the resulting implementation schedules for new licensing issues that arise in the future.

FOR FURTHER INFORMATION CONTACT: Dennis M. Crutchfield, Assistant Director for Safety Assessment, Division of Licensing, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Telephone (301) 492-7492.

SUPPLEMENTARY INFORMATION: In 1977, the Nuclear Regulatory Commission initiated the Systematic Evaluation Program (SEP). Phase I of SEP defined a specific set of safety issues (topics) to be reviewed for operating nuclear power plants. Phase II of SEP was a pilot review of those topics for eleven of the oldest domestic operating reactors. Results have evolved from SEP over the last two years and identified significant experience relative to the safety and evaluation techniques for operating plants.

In 1980, Congress enacted Public Law 96-295 (the NRC Authorization Bill for Fiscal Year 1980). Section 110 of P.L. 96-295 required that the NRC develop a program for the systematic safety evaluation of operating reactors. The program proposal would have extended SEP to an evaluation which required licensees to compare their plant design to the acceptance criteria in the Standard Review Plan (NUREG-0800).* That program was not implemented for operating reactors; the Commission determined, and the Congress agreed, that the scope of the program was too broad to efficiently evaluate the safety of operating reactors. Congress subsequently specified in later Authorization Bills that funds should not be spent to implement that program. However, those activities were useful in that they focused attention on the needs and difficulties associated with the systematic safety evaluation of operating reactors as they relate to a constantly changing technology and increasing scope of regulatory requirements.

Following the TMI-2 accident, the NRC developed the TMI Action Plan (NUREG-0660)* from the safety lessons learned. Two aspects of the TMI Action Plan are particularly significant to the evaluation of the safety of operating plants: (1) it identified a large number of corrective actions to be implemented by operating plants and (2) it initiated the Interim Reliability Evaluation Program (IREP) in which plant-specific probabilistic risk assessment (PRA) studies were to be performed by the staff for several operating reactors to supplement the risk-reliability experience from the Reactor Safety Study (WASH-1400). The licensing actions resulting from TMI have increased the scope of outstanding licensing issues for all operating plants. Similarly, the experience thus far from IREP indicates that there are plant-specific strengths and weaknesses, from a reliability point of view, that warrant further consideration, beyond the deterministically-based issues.

* Copies may be purchased by calling (301) 492-9530 or by writing to the Publications Services Section, Division of Technical Information and Document Control, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, or purchased from the National Technical Information Service, Department of Commerce, 5285 Port Royal Road, Springfield, VA 22161.

One of the most significant conclusions drawn from SEP and IREP is that issues related to safety of operating nuclear power plants can be more effectively and efficiently implemented in an integrated, plant-specific review. In addition, the experience from SEP has served to focus on the set of current licensing criteria which should be evaluated for operating plants and experience from IREP has served to define the methods to conduct a plant-specific probabilistic safety analysis so that consistent, comparable results could be obtained which would enhance an integrated plant safety assessment.

Historically, licensing issues have been evaluated generically, and guidelines for any necessary corrective actions have been applied uniformly to all plants. While this approach has provided an effective means to ensure resolution of these issues, the generic implementation has not given sufficient attention to plant-specific characteristics which have a direct bearing on the appropriateness of the corrective action and the relative importance of the issue in relation to an overall plan for any necessary plant improvements. In some cases, consideration of plant-specific characteristics have identified alternative corrective actions which provide an equivalent or greater measure of safety, often at less cost to the licensee.

Consequently, the NRC has developed the regulatory procedures and attendant policies to conduct integrated assessments for operating power reactors. This approach is called the Integrated Safety Assessment Program (ISAP). In order to ensure the effectiveness of this program, it will be started on a trial basis and the plants to be reviewed have been selected by the NRC Staff from those licensees who indicated an interest to voluntarily participate in such a program.

Based on the results of this trial program, the NRC will decide, in about a year, whether or how this program should be extended to other operating reactors.

IMPLEMENTATION SCHEDULES

To provide a stable environment to conduct ISAP, the Commission has authorized the staff to suspend specific existing implementation schedule requirements for the plants to be reviewed. Each affected licensee will be expected to propose and justify deferral for specific implementation requirements that warrant further evaluation. The associated implementation requirements and other safety issues will be evaluated collectively in an integrated assessment. The staff is only authorized to defer substantive regulatory and other requirements to the extent allowed by the Commission's procedural regulations. Thus, the staff will use the provisions in 10 CFR 50.12 to grant any exemptions.

In addition, any new implementation requirements which evolve late in the course of or following ISAP will be deferred for the plants involved and incorporated in an implementation schedule update, as described below.

The only exceptions will be issues for which the NRC Staff explicitly determines that prompt action is required to protect the health and safety of the public. Such actions include the short-term response to bulletins issued by the Office of Inspection and Enforcement.

SCOPE OF EVALUATION

The scope of ISAP is intended to be as comprehensive as practical. Consequently, it will consist of deterministic, probabilistic, and operating experience evaluations, which will serve to identify specific issues to be addressed in an integrated assessment.

The deterministic review areas, or ISAP topics, will be derived on a plant-specific basis during a screening review with the licensee at the beginning of the program. The issues to be considered are (1) a set of SEP Topics for which the NRC Staff has found significant differences between current licensing criteria and typical design criteria in existence when operating plants were licensed; (2) all pending licensing actions for the plant, including multi-plant actions, TMI Action Plan requirements and plant-specific licensing actions; (3) the unresolved generic issues for which resolution on a plant-specific basis might be expected, and (4) plant improvements proposed by the licensee.

The Commission's Safety Goal Policy published on March 14, 1983, (48 FR 10772), indicates that the quantitative goals and design objectives will not be used in the licensing process during the evaluation period, nor will the policy be interpreted as requiring that licensees or applicants perform a probabilistic analysis; however, the Commission continues to believe that probabilistic analyses provide a valuable adjunct to the deterministic regulatory requirements and enhance engineering judgments, if they are properly performed and applied. Consequently, the Commission believes that a plant-specific probabilistic safety assessment (PSA) should be performed in conjunction with ISAP. The plant-specific PSA will provide a basis for cost/benefit evaluations for the deterministically-based issues and will also identify potential strengths and weaknesses in the plant design and operation which should be considered in an integrated assessment.

An operating experience evaluation will be conducted in parallel with the topic evaluations and plant-specific PSA. This evaluation will be used to identify issues related to significant trends, event precursors, plant management and operation, and maintenance practices. In addition, the operating experience evaluation will provide a diverse perspective for the integrated assessment. The evaluation will consist of an analysis and categorization of reportable events and forced plant shutdowns and an evaluation of overall licensee performance.

EVALUATION PROCESS

The ISAP Topic evaluations and plant-specific PSA will be conducted in parallel. The licensee will initially perform deterministic analyses for the plant-specific set of ISAP Topics by comparing the as-built design of the facility to the current licensing criteria, industry codes and standards, or other appropriate acceptance criteria and also provide risk perspectives for each issue based on the PSA. The staff will review the licensee's analyses and issue safety evaluation reports which identify specific differences from the acceptance criteria and any attendant safety issues which should be considered in the integrated assessment. Schedules for the licensee's analyses and staff evaluation will be established during the screening review to enhance an efficient use of resources.

A PSA would be conducted by the licensee in accordance with an NRC Procedures Guide (NUREG/CR-2815).* The Procedures Guide describes appropriate and consistent methods for (1) initiator definition, (2) application of data, (3) success/failure criteria, (4) analysis, (5) quality control, and (6) documentation and presentation of results. In addition, the NRC Staff will identify methods by which the PSA should address unresolved generic issues; i.e., safety issues for which acceptance criteria do not yet exist. During the screening review, milestones will be established to monitor the progress of the PSA and to ensure appropriate interaction between the NRC Staff and the licensee. The licensee will be expected to use the PSA to identify significant contributors to risk that should be specifically considered in the integrated assessment. For the trial program, the extent and nature of plant-specific probabilistic analyses will be established on a case-by-case basis.

The issues raised in the ISAP Topic evaluations and the PSA, and the operating experience evaluation will be considered collectively in an integrated assessment. Decisions on corrective actions will be based on qualitative assessments of the value and impact of each action. The NRC Staff will present its conclusions regarding the need for or appropriateness of corrective actions proposed by the licensee for each of the identified issues

in a draft report. The draft report will be issued for public comment and peer review. Should the NRC Staff and licensee disagree on the corrective action for any issue, that matter will be resolved in accordance with the Commission's procedures for backfitting requirements.

Following resolution of any comments on the draft report, the NRC Staff will request that the licensee establish and justify implementation schedules for each of the corrective actions and any ongoing analyses that may be necessary to establish appropriate corrective actions. The NRC Staff will judge the adequacy of the proposed implementation schedules based on the technical evaluation of the issues presented in the draft report and issue an implementation plan in a final report.

LICENSING ACTION AND SCHEDULE UPDATES

The final report will serve as the basis and documentation for a license amendment incorporating and formalizing the implementation schedules. The license amendment will also establish procedures to periodically update the implementation schedule.

Any new implementation requirements that arise during or following an ISAP review will be deferred, except for those issues for which the NRC Staff determines that prompt action is required to ensure the health and safety or common defense and security of the public.

The deferred implementation requirements will be evaluated collectively as part of an implementation schedule update. The update evaluations would be conducted periodically, but not more than at five-year intervals. The evaluation would follow the same general course as an ISAP review and would consider a revised PSA, which has been updated to reflect corrective actions and plant

improvements as they are completed. The revised implementation schedule will similarly be incorporated and formalized by a new license amendment.

Dated at Washington, D.C., this 9th day of November 1984.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,
Secretary of the Commission.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICE OF THE
SECRETARY

July 16, 1984

MEMORANDUM FOR: William J. Dircks, Executive Director for Operations

FROM: Samuel J. Chilk, Secretary

SUBJECT: SECY-84-133 - INTEGRATED SAFETY ASSESSMENT PROGRAM (ISAP)

This is to advise you that the Commission (with all Commissioners agreeing) has approved the pilot program for the Integrated Safety Assessment Program. You should revise the Policy Statement as indicated on the attached copies of pages 3 and 4 and return it for publication in the Federal Register following completion of the ongoing 60 day Congressional notification period.
(EDO) (SECY SUSPENSE: 9/15/84 (T))

In approving the pilot program the Commission notes that it has made no commitment beyond the pilot program. The Commission would also like periodic (semi-annual) reports on the progress of the program.

You should also provide the Commission with a report on the status and future of the NREP.
(EDO) (SECY SUSPENSE: 9/1/84)

(Commission action on this item was completed prior to the expiration of Commissioner Gilinsky's term of office).

Attachments:
As Stated

cc: Chairman Palladino
Commissioner Roberts
Commissioner Asselstine
Commissioner Bernthal
OGC
OPE
PDR