NRC PDR

December 28, 1978

SECY-78-616A

COMMISSIONER ACTION

MEMORANDUM FOR: The Commissioners

THRU: Executive Director for Operations

FROM: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

SUBJECT: REVISIONS TO ANNUAL REPORT SECTIONS REGARDING UNRESOLVED SAFETY ISSUES

PURPOSE: To respond to Commission requests regarding SECY-78-616 delineated in memorandum from Samuel J. Chilk to Lee V. Gossick dated December 13, 1978.

Discussion: Mr. Chilk's memorandum of December 13, 1978 delineated a number of actions requested by the Commission as a result of its review of SECY-78-616 regarding reporting the progress of resolution of "Unresolved Safety Issues" in the NRC Annual Report. The staff has revised the proposed Annual Report sections that were included as Enclosure 1 to SECY-78-616 in accordance with the discussions with the Commission. The revised sections are attached. Marginal markings have been included to indicate where revisions have been made. The specific actions taken in response to the numbered requests in Mr. Chilk's December 13, 1978 memorandum are discussed below.

1. Tasks A-12, A-17 and A-36 were added to the list of "Unresolved Safety Issues" and are discussed in the revised Annual Report sections.

2. Explanations along the lines of those given at the December 12, 1978 Commission meeting of why certain generic tasks were not included as "Unresolved Safety Issues" will be provided to the Congress in a NUREG report to accompany the transmittal of the Annual Report.

Contact:
M. Aycock, NRR
49-28041

402 318

45

7907240393

3. The definition of an "Unresolved Safety Issue" has been modified as indicated in the revised Annual Report section. Note that two alternatives are provided that have been suggested by Commissioner Bradford and Commissioner Kennedy. Either alternative is acceptable to the staff.

4. The introductory section of the Annual Report was modified considering the OPE suggestions.

Additional information regarding Task A-30 was provided to Commissioner Bradford's staff by telephone discussions on December 13 and 14, 1978.

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
Revised draft Annual
Report sections

Commissioners' comments should be provided directly to the Office of the Secretary by close of Business Monday, January 8, 1979.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT January 4, 1979 , 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.

DISTRIBUTION
Commissioners
Commission Staff Offices
Exec Dir for Operations
ACRS
AS&LBP
AS&LAP
Secretariat

REVISED

DRAFT ANNUAL REPORT SECTION

ACTION ON TECHNICAL PROBLEMS

NRC actions on technical problems related to nuclear power plant safety can take a number of different forms. They can be (1) specific licensing actions to resolve a problem experienced or identified at an operating reactor, (2) long term research programs, (3) standards development efforts, (4) part of licensing construction permit or operating license reviews, or (5) generic reviews of issues that involve several nuclear power plants.

Items of the first type above that are determined to involve a major reduction in the degree of protection of the public health and safety are reported to Congress quarterly as Abnormal Occurrences (see Chapter 7). Discussions of several additional items involving licensing actions at operating reactors are discussed below under the heading of OTHER ACTIONS.

NRC research programs are discussed in Chapter 11 and the development of regulatory standards is discussed in Chapter 10.

UNRESOLVED SAFETY ISSUES PLAN

Background

In 1977, the Office of Nuclear Reactor Regulation (NRR) instituted a program to define, categorize and manage generic technical activities on a systematic, integrated basis. The initial effort under this program resulted in an identification of 133 generic tasks. These tasks cover a variety of topics, some related to safety, some related to environmental matters and some related to improving the regulatory process.

Subsequent to implementing the NRR program, the Congress in late 1977 acted to amend the Energy Reorganization Act of 1974 to include, among other things, a new Section 210 as follows:

"UNRESOLVED SAFETY ISSUES PLAN"

"Section 210. The Commission shall develop a plan providing for specification and analysis of unresolved safety issues relating to nuclear reactors and shall take such action as may be necessary to implement corrective measures with respect to such issues. Such plan shall be submitted to the Congress on or before January 1, 1978 and progress reports shall be included in the annual report to the Commission thereafter."

402 320

POOR ORIGINAL

In response to this reporting requirement, the NRC provided a report to the Congress (NUREG-0410) in January 1978 describing the generic issues program of the Office of Nuclear Reactor Regulation that had been implemented earlier in 1977. The NRR program provides for the identification of generic issues, the assignment of priorities, the development of detailed Task Action Plans to resolve the issues, projections of dollar and manpower costs, continuing high level management oversight of task progress, and public dissemination of information related to the tasks as they progress. The program described in NUREG-0410 is, however, of considerably broader scope than the "Unresolved Safety Issues Plan" required by Section 210.

As noted above, the NRR program includes other generic tasks of importance to accomplishing the NRC's mission such as tasks for the resolution of environmental issues, for the development of improvements in the reactor licensing process, for consideration of less conservative design criteria or operating limitations in areas where overly conservative requirements may be unnecessarily restrictive or costly, for the maintenance and development of the NRC staff's capabilities to perform independent audit calculations, and for the actual performance of independent audit calculations. This Annual Report section is limited to describing the progress on that portion of the NRR program required to be reported to the Congress by Section 210.

Selection of Issues

The following definition of an "Unresolved Safety Issue" was developed for use in identifying the generic issues in the broader NRR staff program, that should be reported to Congress pursuant to Section 210.

> "An Unresolved Safety Issue is a matter affecting a number of nuclear power plants that poses important questions concerning the adequacy of existing safety requirements [for which a final resolution has not yet been developed]* and that involves conditions not likely to be acceptable over the lifetime of the plants affected."

402 321

---

*The definition has been proposed with and without the bracketed phrase. Either approach is acceptable to the staff.

All of the generic issues reported to the Congress last year in NUREG-0410 were considered as candidates for "Unresolved Safety Issues." A systematic review of these issues was undertaken. As an aid in conducting this review, the topics addressed by these issues were evaluated from the standpoint of their relative contribution to public risk. This risk-based characterization was utilized in conjunction with a substantial body of additional information (e.g., heavy weight was given to issues that resulted from events that have been reported to the Congress as Abnormal Occurrences) to determine which issues met the definition of an "Unresolved Safety Issue." This review resulted in the identification of seventeen "Unresolved Safety Issues." The review process and the rationale for decisions regarding particular issues are described in a separate report, NUREG-0510.

Although the term "Unresolved Safety Issue" has been in use for some time, and the Congress used the term to identify those issues about which it wished to be kept informed, it has been frequently misunderstood. If a generic safety issue (i.e., a safety issue relating to more than one plant) is "unresolved," then how can NRC grant a license to operate a specific nuclear power plant for which that issue is relevant? The answer is that before the license is granted the NRC staff must determine that licensing and operation of the specific plant can continue pending a generic resolution of the issue. The bases for these determinations include one or more of the following: (1) the issue does not apply to or has been resolved for the plant under consideration, (2) interim measures are being required at operating plants pending final resolution of the issue, (3) resolution can reasonably be expected before the plant under consideration begins operation, or (4) the likelihood of occurrence and/or the consequences of an accident scenario for which the issue under study is an important consideration, is small.

The NRC staff's conclusions in this regard are subjected to the scrutiny of the licensing process in individual cases. Specifically, the NRC staff's conclusions on individual applications are reviewed by the Advisory Committee on Reactor Safeguards and are specifically addressed in the public hearing process (see previous section in this Chapter describing the licensing process).

The seventeen generic issues listed below were determined to be "Unresolved Safety Issues." These generic issues are addressed by twenty-two generic tasks in the NRR Program for the Resolution of Generic Issues. The task numbers of the applicable generic tasks are provided in parentheses following the title of each issue. Three of the twenty-two generic tasks addressing these seventeen issues have been completed. Generic Task A-6 was completed and documented in a report, NUREG-0408, "Mark I Containment Short Term Program Safety Evaluation Report," in December 1977; Generic Task A-26 was completed and documented in NUREG-0224, "Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors," in September 1978; and Generic Task A-31 was completed and documented in Regulatory Guide 1.139, "Guidance for Residual Heat Removal," in May 1978.

UNRESOLVED SAFETY ISSUES (APPLICABLE TASK NOS.)

1. Water Hammer - (A-1)
2. Asymmetric Blowdown Loads on the Reactor Coolant System - (A-2)
3. Pressurized Water Reactor Steam Generator Tube Integrity - (A-3, A-4, A-5)
4. BWR Mark I and Mark II Pressure Suppression Containments - (A-6, A-7, A-8, A-39)
5. Anticipated Transients Without Scram - (A-9)
6. BWR Nozzle Cracking - (A-10)
7. Reactor Vessel Materials Toughness - (A-11)
8. Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports - (A-12)
9. System Interactions in Nuclear Power Plants - (A-17)
10. Environmental Qualification of Safety-Related Electrical Equipment - (A-24)
11. Reactor Vessel Pressure Transient Protection - (A-26)
12. Residual Heat Removal Requirements - (A-31)
13. Control of Heavy Loads Near Spent Fuel - (A-36)
14. Seismic Design Criteria - (A-40)
15. Pipe Cracks in Boiling Water Reactors - (A-42)
16. Containment Emergency Sump Reliability - (A-43)
17. Station Blackout - (A-44)

A discussion of each of the "Unresolved Safety Issues" follows.

WATER HAMMER

(GENERIC TASK A-1)

Water hammer events, that is, intense pressure pulses in fluid systems, such as commonly experienced when rapidly closing a water faucet, often occur in nuclear power plant fluid systems. Since 1971, about one hundred incidents involving water hammer in nuclear power reactors have been reported. These water hammer incidents have involved many types of fluid systems including steam generator feed-rings, feedwater and steam supply piping, residual heat removal systems, emergency core cooling systems, containment spray systems, and service water systems. The incidents have been attributed to such causes as the rapid condensation of steam pockets, steam-driven slugs of water, pump start-up with partially empty lines, and rapid valve motions. Most of the damage has been relatively minor, however, there have been several cases of failure or partial failure of system piping.

No water hammer incident has resulted in the release of radioactivity outside of a plant. However, the principal concerns are that water hammer could result in the failure of a pipe in the reactor coolant system or disable a system required to cool the plant after a reactor shutdown.

Means to prevent one particular type of water hammer caused by the rapid condensation of steam in the steam generator feed-rings of some pressurized water reactors are being instituted. Applicants with new steam generator designs are being required to demonstrate through test or analysis that water hammer will not occur in these designs. Plants with steam generators of the top feeding type that are subject to water hammer, are being required to modify the feed-rings and/or test the systems to assure water hammer will not occur. Other actions to correct the specific causes of water hammer identified to-date are also being required.

The NRC staff's review of this safety issue has been incorporated in the NRC Program for Resolution of Generic Issues as Generic Task A-1 as described in a report (NUREG-0410) to Congress submitted in January 1978. The potential for water hammer in various systems is being evaluated and appropriate requirements and systematic review procedures are being developed to ensure that water hammer is given appropriate consideration in all areas of licensing reviews. The task also includes a study of potenti. . water hammer phenomena to .

402 324

aid in the development of design and review procedures. A technical report providing the results of a staff review of water hammer events in nuclear power plants is scheduled for publication in December 1978. Issuance of this report completes a major subtask of Generic Task A-1. The remaining subtasks are expected to be completed in 1980.

402 325

ASYMMETRIC BLOWDOWN LOADS ON THE REACTOR COOLANT SYSTEM
(GENERIC TASK A-2)

In the very unlikely event of a rupture of the primary coolant piping in light water reactors, large nonuniformly distributed loads would be imposed upon the reactor vessel, reactor vessel internals, and other components in the reactor coolant system. These newly identified asymmetric loads, which result from the rapid depressurization of the reactor coolant system, were not considered in the original design of some facilities. The forces associated with a postulated break in the reactor coolant piping near the reactor vessel, for example, could affect the integrity of the reactor vessel supports and reactor pressure vessel internals. A significant degree of failure of the reactor vessel support system, along with impacting the internals, has a potential for (1) damaging systems designed to cool the core following the postulated piping break, (2) affecting the capability of the control rods to function properly, (3) damaging other reactor coolant system components, and (4) causing other ruptures in the initially unbroken reactor coolant system piping loops and attached systems.

The NRC staff's review of this safety issue has been incorporated in the NRC Program for Resolution of Generic Issues as Generic Task A-2. This program including the NRC staff's Task Action Plan for Task A-2 was described in a report (NUREG-0410) to Congress submitted in January 1978.

This issue was originally identified in May 1975 by the Virginia Electric and Power Company in relation to its North Anna Units 1 and 2 nuclear power plants. A survey of all operating PWR reactors was conducted in October 1975 which showed that asymmetric blowdown loads had not been considered in the design of the reactor vessel supports for any operating PWR facility. In June 1976, the NRC staff requested all operating PWR licensees to assess the adequacy of the reactor vessel supports at their facilities with respect to these newly-identified loads.

402 326

Most licensees with Westinghouse plants initially proposed an augmented inservice inspection program (ISI) of the reactor vessel safe-end to end pipe welds in lieu of providing the detailed analysis requested by the NRC staff. Licensees with Combustion Engineering plants submitted a probability study in support of a conclusion that a break at the location in the piping necessary to produce the postulated load had such a low probability of occurrence that no further analysis was necessary. Licensees with Babcock and Wilcox plants took an approach similar to Combustion Engineering licensees.

The NRC staff's review of these proposed alternatives to detailed plant-specific analyses has been completed with the conclusion that the alternative proposals should not be accepted in lieu of the requested analyses.

Accordingly, the NRC staff sent letters on January 25, 1978 to all PWR licensees and applicants stating that an analysis must be undertaken to assess the design adequacy of the reactor vessel supp and other structures to withstand the loads when asymmetric loss-of-coolant accident forces are taken into account. As part of Task A-2, the NRC staff will review and approve analytical models and computer codes developed by reactor vendors to calculate asymmetric blowdown loadings prior to their use by licensees and applicants in plant-specific analyses. In addition, the staff will develop explicit guidelines and acceptance criteria for the asymmetric load analyses and will conduct pipe break probability study to confirm the adequacy of staff decisions related to the continued operation of plants for the interim period while Task A-2, plant-specific analyses, and necessary plant modifications are necessary.

Plant modifications to assure that the postulated loads are accommodated have been implemented late in the construction stage of several plants and have been proposed and are under staff review for some operating plants. For plants still under operating license review, the NRC staff requires that plant-specific analyses be completed and any necessary plant modifications implemented prior to issuance of an operating license. The generic efforts for pressurized water reactors under Task A-2 are currently scheduled for completion in early 1979.

The NRC staff has been investigating this phenomena as it applies to boiling water reactors and has determined that asymmetric loads are also significant and therefore need to be evaluated for these lower pressure systems. The staff is currently developing plans for expanding Task A-2 to resolve this issue for boiling water reactors.

402 327

PRESSURIZED WATER REACTOR STEAM GENERATOR TUBE INTEGRITY (GENERIC TASKS A-3, A-4, A-5)

The heat produced in the reactor at a nuclear power plant is used to convert water into steam which will drive the turbine-generators. In plants employing pressurized water reactors, the primary coolant water which extracts heat by circulating through the reactor core is kept under pressure sufficient to prevent boiling. This high-pressure water passes through tubes around which a secondary coolant (also water) is circulating, under somewhat lower pressure. The water in the secondary system is allowed to boil and produce steam to drive the turbine-generators. The assembly in which the transfer takes place is the steam generator. The tubes within it are an integral part of the primary coolant boundary, keeping the radioactive primary coolant in a closed system and isolated from the environment. The primary concern is the capability of steam generator tubes to maintain their integrity during normal operation and postulated accident conditions. In addition, the requirements for increased steam generator tube inspections and repairs have resulted in significant increases in occupational exposures to workers.

A detailed discussion of the specific problems associated with steam generator tube integrity that were occurring at operating reactors was provided in the 1977 NRC Annual Report, page 95. The information below is provided to supplement and update that information.

Corrosion resulting in steam generator tube wall thinning has been observed in several Westinghouse and Combustion Engineering (CE) plants for a number of years. Major changes in their secondary water treatment process essentially eliminated this form of degradation. Another major corrosion-related phenomenon has also been observed in a number of plants in recent years, resulting from a build-up of support plate corrosion products in the annulus between the tubes and the support plates. This build-up eventually causes a diametral reduction of tubes, called "denting," and deformation of the tube support plates. This phenomenon has resulted in other associated events including stress corrosion cracking, leaks at the tube/support plate intersections and U-bend section cracking of tubes which were highly stressed because of support plate deformation.

402 328

In May 1977, tube denting was discovered at Millstone Unit 2 and the Maine Yankee Atomic Power Plant, both of which had operated exclusively with an all volatile treatment (AVT) of the secondary coolant. It had been thought that this type of treatment might preclude the denting phenomenon from causing significant degradation. The significant developments in Westinghouse and Combustion Engineering steam generators, since June 1977, were the following:

- Continued tube denting at Indian Point Unit 2, San Onofre Unit 1, Surry Units 1 and 2, Turkey Point Units 3 and 4, and lesser amounts of denting at a number of other Westinghouse designed reactors. Steam generator replacement is planned for early 1979 or 1980 at Surry Units 1 and 2. Replacement or retubing is also being considered for Turkey Point Units 3 and 4. In the interim, the units are operating under restrictions imposed by the NRC.
- Discovery of support plate cracking (related to denting) at Indian Point Unit 2 and San Onofre Unit 1.
- Removal of several tubes and a section of support plate at Indian Point Unit 2 to investigate the potential for steam generator cleaning revealed continued active corrosion of the support plate.
- Continuation of tube denting at Millstone Unit 2 and Maine Yankee and discovery of denting in St. Lucie 1. Millstone Unit 2, Maine Yankee, and Arkansas Nuclear One Unit 2 have removed lugs and portions of the solid rim in the uppermost support plates to reduce the susceptibility of the plates to denting-related cracks (CE designs).
- Palisades Nuclear Power Station is sleeving degraded tubes instead of plugging them. This process restores the structural integrity of the tubes while keeping them in service (CE design).

Steam generator tube degradation in Babcock and Wilcox (B&W) steam generators has been limited to the Oconee Nuclear Plant where the first tube leak occurred in July 1976. In the last quarter of 1976 and the first quarter in 1977, there was a total of seven plant shutdowns to plug leaking tubes in the three Oconee units. To-date, 14 tube leaks, all at the Oconee units, have occurred in B&W steam generators. The majority of these leaking tubes were located adjacent to the open inspection lane. Laboratory examination of removed defective tubes indicated that the tube failures were caused by the propagation of circumferential fatigue cracks, of unknown origin, by flow-induced vibration.

402 329

The significant developments in B&W steam generators, since May 1977, were the following:

- Continued tube leaks at the Oconee units.
- Initiation of a demonstration tube sleeving program by Duke Power Company at the Oconee units. The tube sleeves will not serve as part of the primary coolant boundary but will be installed to change the vibrational characteristics of the tubes and decrease the dynamic stresses and the susceptibility of the tubes to fatigue cracking.

Following inspections by licensees of their steam generators and the completion of any necessary repair programs, the NRC usually must approve or concur in the restart of each of the severely affected facilities. To-date, the units severely affected by the tube denting have completed inspection and repair programs and received NRC approval for operation for limited time periods. Safe operation is assured by the imposition of strict conditions on licensed operation, requiring the plugging of affected tubes and restricting allowable leak rates during operation.

As the NRC staff continues to closely monitor, evaluate, and approve the acceptability of continued operation of plants experiencing steam generator tube problems, it has undertaken a number of generic reviews and studies as part of three generic tasks in the NRC Program for the Resolution of Generic Issues; specifically, Generic Tasks A-3, A-4, and A-5 each directed at the particular problems of Westinghouse, Combustion Engineering and Babcock and Wilcox plants, respectively.

Under these tasks generic studies will be conducted to (1) evaluate inservice inspection results from operating reactors, (2) evaluate the consequences of tube failures under postulated accident conditions, (3) evaluate tube structural integrity, (4) establish tube plugging criteria based on new information, (5) define the requirements for monitoring secondary coolant chemistry, (6) evaluate inservice inspection methods, and (7) review design improvements proposed for new plants. These studies will be used to revise current NRC staff requirements and guidance regarding these subjects. In addition, under Task A-3, the NRC staff will review and evaluate the first proposed steam generator replacement operation to establish acceptance criteria and guidance on a generic basis for use in the review of subsequent replacement operations. These generic tasks are currently scheduled to be completed in early 1980.

BWR MARK I AND MARK II PRESSURE SUPPRESSION CONTAINMENTS
(GENERIC TASKS A-6, A-7, A-8, A-39)

In the course of performing large scale testing of an advanced design pressure-suppression containment (Mark III), and during in-plant testing of Mark I containments, new suppression pool hydrodynamic loads were identified which had not explicitly been included in the original Mark I or Mark II containment design basis. These additional loads result from dynamic effects of drywell air and steam being rapidly forced into the suppression pool (torus) during a postulated LOCA and from suppression pool response to various modes of safety relief valve (SRV) operation generally associated with plant transient operating conditions. Since these new hydrodynamic loads had not been explicitly considered in the original design of the Mark I and Mark II containments, the NRC staff determined that a detailed reevaluation of these containment system designs was required.

As a result of the need for this reevaluation the affected utilities formed ad hoc Mark I and Mark II Owners' Groups and each has engaged the General Electric Company as its program manager. Both Owners' Groups developed two-phase programs consisting of a short-term program and a long-term program for resolution of the pool dynamic concerns for their respective containment designs. The Owners' Groups' programs consist of among other things, a number of comprehensive experimental and analytical programs to establish generic pool dynamic loads, load combinations and design criteria.

The NRC staff has identified and initiated a number of generic tasks to review and evaluate the results of the Mark I and Mark II Owner's Group short-term and long-term programs to develop technical positions for use in licensing actions on individual plants utilizing the Mark I and Mark II containment designs. These generic tasks are included in the NRC Program for Resolution of Generic Issues (described in NUREG-0410 as noted above). Specifically, they are Task A-6, Mark I Short-Term Program; Task A-7, Mark I Long-Term Program; Task A-8, Mark II Containment Program; Task A-39, Determination of Safety Relief Valve (SRV) Pool Dynamic Loads and Temperature Limits for BWR Containments.

The objectives of the Mark I Short-Term Program were: (1) to examine the containment system of each BWR facility with a Mark I containment design to verify that

402 331

it would maintain its integrity and functional capability when subjected to the most probable hydrodynamic loads induced by a postulated design basis loss-of-coolant accident; and (2) to verify that licensed Mark I BWR facilities may continue to operate safely, without undue risk to the health and safety of the public, while a methodical, comprehensive Long-Term Program is conducted. The NRC determined that, for the Short-Term Program, "maintenance of containment integrity and function" would be adequately assured if a safety factor to failure of at least two were demonstrated to exist for the weakest structural or mechanical component in the Mark I containment system (i.e., if the calculated stresses in all components of the affected containment structure were shown to be less than one-half the stress which would cause the component to lose its structural integrity). The NRC concluded that the objectives of the Short-Term Program had been satisfied and documented the basis for this conclusion in the "Mark I Containment Short-Term Program Safety Evaluation Report," NUREG-0408, dated December 1977.(i.e., Task A-6 was completed in December 1977).

The objectives of the Mark I Long-Term Program are: (1) to establish design basis loads that are appropriate for the anticipated life of each Mark I BWR facility, and (2) to restore the original intended design safety margins for each Mark I containment system. The Mark I Long-Term Program consists of a series of major tasks and subtasks which are designed to provide a detailed basis for hydrodynamic load definition and the methodology and acceptance criteria for the structural assessments. The generic aspects of the Mark I Long-Term Program will be described in a Plant Unique Analysis Applications Guide, scheduled to be completed in October 1978, and in the Load Definition Report, scheduled to be completed in December of 1978. Subsequently, each utility with a Mark I plant will perform a plant-unique analysis using approved load definition and structural analysis techniques to demonstrate conformance with the Mark I Long-Term Program structural acceptance criteria. These analyses are currently scheduled for completion in October 1979.

The scheduled completion date for the Mark I Long-Term Program (Task A-7) including the issuance of license amendments and the implementation of any plant modifications necessary to satisfy the Mark I Long-Term Program structural acceptance criteria, is December 1980. In recognition of this schedule, a number of facilities are adopting their own schedules to implement anticipated plant modifications and minimize the potential for extended plant outages or unscheduled outages.

The objective of the NRC staff's efforts under Generic Task A-8 related to the Mark II Short-Term Program (STP) was to review and evaluate the pool dynamic loads associated with a postulated large loss-of-coolant accident proposed by the Mark II Owner's Group to determine their acceptability for use in plant unique analyses. The Mark II Short-Term Program was completed in October 1978 and documented in NUREG-0487, "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria." With regard to the Mark II Long-Term Program (LTP), the NRC staff will evaluate the results of the Mark II confirmatory experimental and analytical programs to assess the margin for selected loads. The Mark II Long-Term Program is currently scheduled for completion in October 1980.

Under Generic Task A-39, the NRC staff will review and evaluate the results of the Mark I and Mark II Owners' Group's experimental and analytical programs to establish and justify the safety relief valve-related pool dynamic loads for BWR Mark I and Mark II containment designs. The results of Generic Task A-39 will be an integral part of the final acceptability of the Mark I and Mark II pressure suppression containment designs. This generic task is currently scheduled for completion in December 1979. An interim assessment of multiple-consecutive SRV discharges is currently being performed for the operating Mark I facilities to support deferral of this issue until the completion of the Mark I Long-Term Program. The review of these assessments is scheduled for completion in November 1978.

ANTICIPATED TRANSIENTS WITHOUT SCRAM
(GENERIC TASK A-9)

Nuclear plants have safety and control systems to limit the consequences of temporary abnormal operating conditions or "anticipated transients." Some deviations from normal operating conditions may be minor; others, occurring less frequently, may impose significant demands on plant equipment. In some anticipated transients, rapidly shutting down the nuclear reaction (initiating a "scram"), and thus rapidly reducing the generation of heat in the reactor core, is an important safety measure. If there were a potentially severe "anticipated transient" and the reactor shutdown system did not "scram" as desired, then an "anticipated-transient-without-scram," or ATWS, would have occurred.

This issue has been discussed throughout the nuclear industry for a number of years. Historically, the regulatory staff has excluded very low probability events from the design basis. At issue in the ATWS discussions is whether or not the probability of an ATWS event is sufficiently low to warrant the continuance of the current staff practice with regard to ATWS, i.e., continued exclusion from the design basis for nuclear power plants because of its low probability.

Because of the perceived potential for serious consequences resulting from ATWS events, a number of studies have been undertaken to assess the probabilities and consequences of such events. These studies have been performed by vendors, utility groups, and by the AEC and NRC regulatory staff. The ATWS issue was incorporated in the NRC Program for Resolution of Generic Issues (described in NUREG-0410 as noted above) as Generic Task A-9.

In September 1973, the then-AEC staff published WASH-1270, "Technical Report on Anticipated Transients Without Scram for Water Cooled Power Reactors," which set forth staff "acceptance criteria" to protect against ATWS events. During the two-year period following publication of the staff report, each of the four reactor manufacturers submitted analyses and supporting information on ATWS which was reviewed by the NRC staff and addressed in four status reports published in December 1975. The staff reports evaluated the information for conformance to the WASH-1270 criteria and noted where design changes and additional analyses were required.

The vendors and owners have questioned whether the NRC staff's requirements are necessary and justified. The industry contends that the probability of an ATWS event is significantly less than estimated by the NRC staff and so low as to make ATWS events minor safety concerns in light water reactor operations.

Because of the continuing controversy over the NRC staff position since its publication in WASH-1270, a staff review and evaluation of all the information available on the subject of ATWS, and in particular, the material developed subsequent to the publication of the staff status reports referred to above, was undertaken in the latter part of 1977 and early 1978. A report, NUREG-0460, was published in April 1978 providing the results of this review and evaluation.

It was concluded in NUREG-0460 that considering the expected frequency of transients, the reliability of current reactor scram systems necessary to meet the safety objectives has not been demonstrated and may well have not been attained. NUREG-0460 recommended that means of mitigating the consequences of ATWS events be provided in plant designs.

The recommendations presented in NUREG-0460 have been criticized by industry and some members of the staff as unnecessarily conservative and therefore too costly. The staff is now evaluating alternative means of reducing the probability or consequences of ATWS events, other than that recommended in NUREG-046 . The effectiveness, cost and other factors such as the effect on the licensing process of these alternatives is being evaluated. Based on this evaluation, the staff will recommend to the Commission the alternatives which provide the best balance between safety and cost for new designs, plants under construction and operating plants. The staff expects to provide its recommendations to the Commission in early 1979.

BWR NOZZLE CRACKING
(GENERIC TASK A-10)

Over the last several years, inspections at 20 of the 23 boiling water reactor (BWR) plants licensed for operation in the U.S. have disclosed some degree of cracking in the feedwater nozzles of the reactor vessel at all but two facilities. The exceptions were a plant with less than one year of operation and a plant with welded nozzle thermal sleeves. The three other facilities have not yet accumulated significant operating time and have not yet been inspected.

The feedwater nozzles, part of the "pressure vessel," are an integral part of the primary pressure boundary of the reactor coolant system and the second barrier (after the fuel cladding) to the release of radioactive fission products. All of the repaired BWR feedwater nozzles met the ASME pressure vessel code limits, however, and no immediate action was necessary. Because relatively small amounts of base metal have been removed by repair operations, there has been no significant reduction in safety margins. Several plants have removed the stainless steel nozzle cladding as a means of eliminating crack initiation since the clad thickness was not necessary to meet code reinforcement requirements. Nevertheless, the cracking is potentially serious because:

- Excessive crack growth could lead to impairment of pressure vessel safety margins requiring more complicated repair work than simple grinding.
- The design safety margin could be reduced by excessive removal of base metal.
- The exposure to radiation of the personnel performing inspection and repair tasks can be considerable.
- The repair of these kinds of cracks can result in considerable shutdown time at the plant affected.

The reactor vendor (the General Electric Company) and the NRC have concluded from their respective studies that the cracking is caused by fluctuations or "cycling" of the temperature on the inside surface of the nozzles; that the stainless steel cladding exhibited less resistance to crack initiation than the underlying low-alloy steel; and that, after initiation in the stainless steel cladding, cracks can be propagated by operational startup and shutdown cycles or other operationally-induced transients. The vendor has performed extensive analysis and testing to confirm the suspected cause of the cracking and to uncover possible long-term solutions: a newly designed sleeve, removal of the

402 336

stainless steel cladding, reduction of the temperature differential at the nozzle, or some combination of these. The licensees involved have increased the number and extent of inspections of feedwater nozzles, with careful repair and reinspection where cracks were found. The vendor advised these licensees to closely monitor startup and shutdown procedures in an effort to substantially reduce the time during which cold feedwater is being injected into the hot pressure vessel.

In a closely related area, the NRC was informed in March 1977 by the General Electric Company that a crack had been found in the nozzle of the "control rod drive (CRD) return line" in a reactor vessel in a foreign country. The CRD return line nozzles are the openings in BWR pressure vessels through which the high pressure water in excess of that needed to operate and cool the CRDs is returned to the pressure vessel. Later in March, the Philadelphia Electric Company reported that similar cracking had been found in the CRD return line nozzle at its Peach Bottom Atomic Power Station, Unit 3. The cracks resembled those found in the feedwater nozzles and seemed to be the result of the same kind of cyclic thermal stresses that were causing feedwater nozzle cracks. Both the foreign reactor and the Peach Bottom Unit 3 reactor are representative of a small number of BWRs which do not have a thermal sleeve in the CRD return line nozzle.

The licensee removed the cracks in the Peach Bottom CRD nozzle by grinding out the cracked area, the maximum crack depth being 7/8-inch, and returned the unit to operation with the CRD return line "valved out" and with the flow and pressure in the CRD hydraulic system modified.

Inspection of other CRD return line nozzles which incorporated thermal sleeves indicated that these sleeves may not be effective in preventing this cracking phenomenon. The Georgia Power Company found a crack in the CRD return line nozzle at its Hatch Plant, Unit 1, which did have a thermal sleeve. (The crack was removed, the nozzle capped, and the return line rerouted to the reactor water cleanup system.)

The NRC staff efforts relate to the resolution of these two similar issues regarding nozzle cracking in boiling water reactors were consolidated into a single staff effort, Generic Task A-10, in 1977. Under Generic Task A-10, the staff issued interim guidance to operating plants in a report entitled, "Interim Technical Report on BWR Feedwater and Control Rod Drive Return Line Nozzle Cracking" in July 1977. The staff is often requiring inservice inspection using liquid penetrant examinations at operating reactors in accordance with the frequency, procedures and acceptance criteria described in the above report.

Additional efforts under Generic Task A-10 include following and reviewing advancements in (1) the development and testing of effective feedwater nozzle thermal sleeves and spargers, (2) life-cycle testing of certain CRD system valves, (3) the development of various feedwater system and CRD system modifications, and (4) the development of viable ultrasonic system techniques by the nuclear industry to allow reliable and consistent early determination of cracking from positions exterior to the reactor vessel.

Generic Task A-10 is scheduled for completion in late 1979.

REACTOR VESSEL MATERIALS TOUGHNESS
(GENERIC TASK A-11)

Resistance to brittle fracture, a rapidly propogating catastrophic failure mode, for a component containing flaws is described quantitatively by a material property generally denoted as fracture toughness. This resistance to fracture, or fracture toughness, has different values and characteristics depending upon the material being considered. For nuclear reactor pressure vessel steels, three considerations are important. First, fracture toughness increases with increasing temperature. Second, fracture toughness decreases with increasing load rates. Third, fracture toughness decreases with neutron irradiation.

In recognition of these considerations, power reactors are operated within restrictions imposed by the Technical Specifications on the pressure during heatup and cooldown operations. These restrictions assure that the reactor vessel will not be subjected to that combination of pressure and temperature that could cause brittle fracture of the vessel if significant flaws in the vessel material exist. The effect of neutron radiation on the fracture toughness of the vessel material is accounted for in developing and revising these Technical Specification limitations over the life of the plant.

For the service times and operating conditions typical of current operating plants, reactor vessel fracture toughness properties provide adequate margins of safety against vessel failure. Further, for most plants the vessel material properties are such that adequate fracture toughness can be maintained over the life of the plants. However, results from reactor vessel surveillance program indicate that up to 20 older operating pressurized water reactors were fabricated with materials that will have marginal toughness after comparatively short periods of operation.

The objective of Task A-11 is to evaluate material degradation mechanisms resulting from neutron irradiation and determine appropriate licensing criteria and corrective action for low toughness reactor vessel materials in these

402 339

currently licensed plants. Task A-11 is currently scheduled for completion in July 1979. This completion date is well in advance of the date needed to assure that adequate fracture toughness is maintained in these older plants.

FRACTURE TOUGHNESS AND POTENTIAL FOR LAMELLAR TEARING OF PWR STEAM GENERATOR AND REACTOR COOLANT PUMP SUPPORTS (GENERIC TASK A-12)

During the course of licensing review for a specific Pressurized Water Reactor (PWR) a number of questions were raised as to (1) the adequacy of the fracture toughness properties of the material used to fabricate the reactor coolant pump and steam generator supports and (2) the potential for failure due to lamellar tearing of these same supports. The safety concern is that, although these supports are designed for worst-case accident conditions, poor fracture toughness or lamellar tearing could cause the supports to fail if severely loaded during such accidents. Support failure could conceivably impair the effectiveness of systems designed to mitigate the consequences of the accident. An example of a postulated event sequence of potential concern would be a large pipe break in the reactor coolant system which severely loads the supports, followed by a support failure of sufficient magnitude that a major component such as a steam generator is severely displaced resulting in failure of the emergency core cooling system piping which is needed to provide cooling water to the core.

Two different steel specifications (ASTM A36-70a and ASTM A572-70a) covered most of the material used for the supports of the PWR in question. To address the fracture toughness question (lamellar tearing is discussed separately below) tests not originally specified and not in the relevant ASTM specifications were made on those heats of steel for which excess material was available. The toughness of the A36 steel was found to be adequate, but the toughness of the A572 steel was relatively poor at an operating temperature of 80°F. In the case of the PWR in question, the applicant agreed to a license condition which stated that he would raise the temperature of the ASTM A572 beams in the steam generator supports to a minimum temperature of 225°F prior to reactor coolant system pressurization to levels above 1000 psig, assuring adequate toughness in the event of an accident. Auxiliary electrical heat will be used to supplement the heat derived from the reactor coolant loop to obtain the required operating temperature of the support materials.

Because similar materials and designs have been used in other plants and therefore similar problems may exist, we have incorporated the review of this issue in the NRC Program for Resolution of Generic Issues as Generic Task A-12.

A consultant was engaged to reassess the fracture toughness of the steam generator and reactor coolant pump support materials for all operating PWR plants and those in the later stages of operating license review.

The staff has completed a review of the materials utilized in the supports of 34 potentially affected PWRs. Based on the consultant's preliminary evaluation, we have determined that there are approximately 15-20 plants whose supports have questionable toughness. We expect that these plants may be required to utilize inservice inspection or auxiliary heating of adequate toughness properties cannot be demonstrated.

Upon completion of our generic study, we will document the generic phase of the fracture toughness program and will begin to implement the results on a plant-specific basis. The generic solution will result in changes to the Standard Review Plan to incorporate the lessons learned for use in future license reviews.

The staff has concluded that continued operation (and licensing) of PWRs is justified pending completion of this task and implementation of the task results. Support failure is not expected to occur except under the unlikely combination of:

(1) The occurrence of an initiating event (e.g., a large pipe break) which has been determined to be of low probability (normal operating stresses on piping are very low).

(2) The existence of non-redundant and critical support structural member(s) with low fracture toughness (many supports contain redundant members).

(3) The existence of support structural members at operating temperatures low enough that the fracture toughness of the support material is reduced to the level that brittle failure could occur if a large flaw existed.

(4) The existence of a flaw of such large size that the stresses imparted during the initiating event could cause the flaw to rapidly propagate resulting in brittle failure of the member(s).

402 342

The second potential concern (i.e., lamellar tearing[1/]) may also be a problem in those support structures similar in design to the aforementioned PWR. However, continued operation of PWRs during our continuing generic review of this concern is acceptable, based on the fact that a review to date of approximately 400 related technical documents revealed only one instance of known failure from lamellar tearing. This failure occurred in often-stressed truck brakes. In addition, the factors considered above for the fracture toughness concern, such as low stresses during normal operation and the low probability of an initiating event equally apply to this concern.

The generic fracture toughness program is expected to be completed in August 1979. The lamellar tearing evaluation is a longer term effort and is expected to be completed in 1981.

[1/] Lamellar tearing is a cracking phenomenon which occurs beneath welds and is principally found in rolled steel plate fabrications. The tearing always lies within the parent plate, often outside the transformed (visible) heat-affected zone (HAZ) and is generally parallel to the weld fusion boundary. Lamellar tearing occurs at certain critical joints usually within large welded structures involving a high degree of stiffness and restraint. Restraint may be defined as a restriction of the movement of the various joint components that would normally occur as a result of expansion and contraction of weld metal and adjacent regions during welding ("Lamellar Tearing in Welded Steel Fabrication", The Welding Institute).

SYSTEMS INTERACTIONS IN NUCLEAR POWER PLANTS

(GENERIC TASK A-17)

In November 1974 the Advisory Committee on Reactor Safeguards requested the NRC staff to give attention to the evaluation of safety systems from a multi-disciplinary point of view to identify potentially undesirable interactions between plant systems. The concern arises because the design and analysis of systems is frequently assigned to teams with functional engineering specialties such as civil, electrical, mechanical, or nuclear. The question is whether the work of these functional specialists is integrated in the design and analysis so as to identify adverse interactions between and among systems.

These adverse events might occur because designers might not, for example, assure that redundancy and independence of safety systems are provided under all conditions of operation where redundancy and independence is required because the functional teams may not be adequately coordinated. Simply stated: the left hand may not know or understand what the right hand is doing in all cases where it is necessary for the hands to be coordinated.

The NRC staff believes that its current review procedures and safety criteria provide reasonable assurance that an acceptable level of redundancy and independence is provided for systems that are required for safety. Nonetheless, in mid-1977 this task (Task A-17) was initiated to investigate systems interaction from the point of view of confirming that our present procedures acceptably account for potentially undesirable interactions between and among systems.

The NRC staff's current review procedures assign primary responsibility for various technical areas and safety systems to specified technical review organizational units and assign secondary responsibility to other units where there is a functional or interdisciplinary relationship. Designers follow somewhat similar procedures and provide for interdisciplinary reviews and analyses of systems. Task A-17 will provide an independent investigation of safety functions and systems required to perform these functions in order to assess the adequacy of current review procedures. This investigation will be conducted by Sandia Laboratories under contract assistance to the NRC staff.

402 344

The contract effort, Phase I of the task, began in May 1978 and is expected to be completed in September 1979. The Phase I investigation is structured to identify where interactions are possible between and among systems where these interactions have the potential of negating or seriously degrading the performance of safety functions. The investigation will then proceed to identify where our review procedures may not have properly accounted for these interactions. Finally, based on a determination of the overall significance to safety, in a follow-on Phase II of the task, specific corrective measures will be taken in areas where the investigation shows a need.

As noted above, the NRC staff believes that its review procedures and acceptance criteria currently provide reasonable assurance that an acceptable level of system redundancy and independence is provided in plant designs and this task is expected to confirm this belief. Nonetheless, because adverse systems interactions are potentially of large significance to plant safety this issue has been included as an "Unresolved Safety Issue." If no significant system interactions are identified in the Phase I investigation described above, as is expected, this issue will not be treated in subsequent reports as an "Unresolved Safety Issue."

ENVIRONMENTAL QUALIFICATION OF

SAFETY-RELATED ELECTRICAL EQUIPMENT

(GENERIC TASK A-24)

Despite the conservative design, construction and operating practices and quality assurance measures required for nuclear power plants, safety systems are installed at nuclear plants to mitigate the consequences of postulated accidents. Some postulated accidents could create severe environmental conditions inside of containment. The most limiting of these accidents are high energy pipe breaks in the reactor coolant system piping or in a main steam line. In either of these cases, the release of hot pressurized water and steam to the containment creates a high temperature environment (250 to 400°F) at high humidity (including steam) and pressure (as high as about 50 psig). For some applications, chemicals are added for fission product removal to the containment sprays that are used to reduce the pressure in the containment. Additionally, some electrical equipment is predicted to be submerged following a large pipe break. Thus, the safety equipment is exposed to such environmental conditions and needs to remain operable during this period, as well as for the long-term post-accident period.

The NRC requires that electrical equipment in safety systems, principally the emergency core cooling system and containment isolation and cleanup systems, be environmentally qualified to assure that this equipment will perform its required function in the environment associated with such severe accidents. Specific electrical equipment of concern during postulated accident conditions includes (1) the instrumentation needed to initiate the safety systems and provide diagnostic information to the plant operators (e.g., electrical penetrations into containment, any electrical connectors to cabling which transmits signals, and the instruments themselves), (2) control power to motor operators for certain valves (e.g., ECCS and containment isolation valves located inside containment), and (3) fan cooler motors for those plants that utilize fan coolers for containment heat removal.

The current NRC safety review process for nuclear power plants includes criteria related to the qualification of certain electrical equipment. These criteria require that electrical equipment important to safety must be qualified to function in the environment that might result from various accident conditions. Although such criteria have been applied to varying degrees since the early days of commercial nuclear power, the details of these criteria have been more clearly defined over the years.

These clarifications of the criteria have raised some questions as to:

(1) the degree to which electrical equipment used in older plant designs (those now operating) is capable of withstanding the environmental conditions (pressure, temperature, humidity, steam, chemicals, vibration, and radiation) of various accident conditions under which it must function (i.e., the "qualification of equipment" in these older plants), and

(2) the adequacy of test or analyses conducted for electrical equipment in newer plants to "qualify" such equipment as capable of withstanding the conditions of the environment created by various accidents during which the equipment must function (i.e., the "adequacy" of qualification tests).

With regard to older plants, the following actions have taken place in recent months.

As a result of a Sandia testing program being conducted for the Office of Nuclear Regulatory Research, a generic safety concern with the adequacy of environmental qualification of certain electrical equipment was identified. This issue was highlighted by a November 4, 1977 petition from the Union of Concerned Scientists which requested immediate action regarding operating power reactors and licensing actions for other proposed plants.

Subsequen NRC staff investigations in response to this issue have led, as of June 1978, to seven plant shutdowns for corrective action and extended outages for two other plants to make modifications. These actions were for the most part a result of a lack of conclusive information regarding the qualification of certain safety equipment.

Having identified problems associated with qualification of electrical equipment, the NRC conveyed its information to the licensees of all operating reactor facilities through an Inspection and Enforcement Circular which was issued on May 31, 1978. The purpose of this Circular was to ensure that the knowledge gained by the NRC staff and the lessons learned would be appropriately factored into future actions. The NRC staff also has initiated an augmented inspection effort as part of the normal NRC activities. This effort will concentrate on the inspection of installed safety-related electrical equipment and on an audit of the records for environmental qualification.

Additionally, a review of the environmental qualification of safety-related electrical equipment has been initiated for 11 operating reactor facilities in the Systematic Evaluation Program (SEP). (See Chapter 7, "Abnormal Occurrences - 1978.")

With regard to the second question above, the NRC staff has worked with the industry to develop standards for equipment qualification and documentation which would assure the high level of equipment reliability required for nuclear applications. This effort has culminated in the development of IEEE Std. 323, "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations." This standard and its ancillary standards have provided the focal point for the development of environmental qualification requirements in recent years.

IEEE Std. 323 was first issued as a trial use standard (IEEE Std. 323-1971) in 1971 and later, after substantial revision, as a final standard (IEEE Std. 323-1974) in 1974. Both versions of the Standard set forth basic requirements for environmental qualification of electrical equipment but do not provide specific details for implementation of these requirements. Specific qualification techniques have

402 348

POOR ORIGINAL

been reviewed and approved by the NRC staff on a case-by-case basis as a part of individual licensing actions. These licensing actions include initial construction permit and operating license application reviews and requalification actions for operating reactors where documentation of the initial qualification was not available.

The evolutionary nature of the process of developing environmental qualification requirements and the case-by-case implementation of these requirements has resulted in a diversity of methods in use and different levels of documentation of the extent to which equipment is qualified.

Several aspects of equipment qualification are being pursued at this time by the NRC staff and the nuclear industry on a generic basis to achieve a more uniform implementation of the general qualification requirements established in IEEE Std. 323-1974. One such activity is the development of interim NRC staff positions regarding how the requirements of IEEE Standard 323-1974 can be met. This activity is a part of Generic Task A-24, "Environmental Qualification of Safety-Related Electrical Equipment," in the NRC Program for the Resolution of Generic Issues and is scheduled for completion in .

Further efforts under Generic Task A-24 involve the review of the environmental qualification programs of reactor vendors and architect/engineers as a basis for qualifying safety-related electrical equipment to the requirements of IEEE-Standard 323-1974. Performing these reviews on a generic basis rather than on case-by-case licensing reviews will provide resource savings for the NRC staff and the industry. This follow-on portion of the generic task will be scheduled following completion of the development of the interim NRC staff positions referred to above.

REACTOR VESSEL PRESSURE TRANSIENT PROTECTION
(GENERIC TASK A-26)

Over the past several years, incidents identified as pressure transients have occurred in pressurized water reactors (PWR). To-date, there have been thirty-three such events. Half of these events occurred before the plant achieved initial criticality (i.e., before initial operation of the reactor). The majority occurred during startup or shutdown operations. All of the pressure transients were such that fracture mechanics and fatigue calculations indicate that the reactor vessels were not damaged and continued operation of these vessels was acceptable. Nevertheless, the staff concluded that appropriate regulatory actions were necessary, (1) to reduce the frequency of pressure transient events and (2) to provide equipment which would restrict future transients to acceptable pressures. This action was necessary because reactor vessel safety margins would be reduced during the lifetime of the vessel due to neutron irradiation causing reduced material toughness.

The NRC staff's review of this safety issue was incorporated in the NRC Program for Resolution of Generic Issues as Generic Task A-26. The final report, NUREG-0224, "Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors," was issued in September 1978. This previously "Unresolved Safety Issue" has been resolved.

Upgraded procedural controls were implemented at operating PWR facilities which significantly reduced the occurrence of pressure transient events. The few events which have occurred were not significant and were of the type that will be precluded by equipment changes.

The majority of the equipment changes implemented at operating PWR facilities involve the addition of a second lower set point on existing power operated relief valves, the addition of new spring-loaded relief valves, or modifications to allow use of existing spring-loaded relief valves. A few newly licensed facilities must complete similar design changes by their first refueling shutdown. The extended equipment implementation schedule for new facilities was based upon the reduced frequency of occurrence of pressure transient events due to improved procedural controls and the large safety margins for new pressure vessels.

RESIDUAL HEAT REMOVAL SHUTDOWN REQUIREMENTS
(GENERIC TASK A-31)

The safe shutdown of a nuclear power plant following an accident not related to a loss-of-coolant accident (LOCA) has been typically interpreted as achieving a hot-standby condition (i.e., the reactor is shutdown, but system temperature and pressure are still at or near normal operating values). Consequently, considerable emphasis has been placed on the hot-standby condition of a power plant in case of an accident or abnormal occurrence. A similar degree of emphasis has been placed on long-term cooling, which is typically achieved by the residual heat removal (RHR) system. The RHR system starts to operate when the reactor coolant pressure and temperature are substantially lower than their hot-standby condition values.

Even though it may generally be considered safe to maintain a reactor in a hot-standby condition for a long time, experience shows that there have been events that required eventual cooldown and long-term cooling until the reactor coolant system was cold enough to perform inspection and repairs. It is therefore obvious that the ability to transfer heat from the reactor to the environment after a shutdown is an important safety function for both PWRs and BWRs. Consequently, it is essential that a power plant have the capability to go from hot-standby to cold-shutdown conditions (when this is determined to be the safest course of action) under any accident conditions.

This issue was adopted as a Category A issue and designated as Task A-31, "RHR Shutdown Requirements" in 1977. It was described in the NRC Report to Congress, NUREG-0410, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," issued on January 1, 1978.

...rdance with the Task Action Plan for this task, the staff's views on requirements for residual heat removal systems were translated into proposed changes to Standard Review Plan Section 5.4.7. These proposals were considered by the Regulatory Requirements Review Committee (RRRC) during its 71st meeting on January 31, 1978.

The RRRC recommended approval of the proposed changes and further recommended that (1) the changes be applied on a case-by-case basis to all operating reactors and all other plants (custom or standard) for which the issuance of the operating license is expected before January 1, 1979, and (2) the changes be backfitted to all plants (custom or standard) for which construction permit or preliminary design approval applications were docketed before January 1, 1978, and for which the operating license issuance is expected after January 1, 1979. These recommendations were approved by the Director, NRR and are being implemented.

Subsequently, the staff positions on design requirements for residual heat removal systems were incorporated into Regulatory Guide 1.139, "Guidance for Residual Heat Removal", which was issued for public comment in May 1978. Comments were received during the latter part of 1978 and it is expected that this Regulatory Guide can be issued in its final form in late 1979 or early 1980.

CONTROL OF HEAVY LOADS NEAR SPENT FUEL
(GENERIC TASK A-36)

Overhead handling systems (cranes) are used to lift heavy objects in the vicinity of spent fuel in PWRs and BWRs. If a heavy object, e.g., a spent fuel shipping cask or shielding block, were to fall or tip onto spent fuel in the storage pool or the reactor core during refueling and damage the fuel, there could be a release of radioactivity to the environment and a potential for radiation over-exposures to inplant personnel. If the dropped object is large, and is assumed to drop on fuel containing a large amount of fission products with minimal decay time, calculated offsite doses could exceed the siting guideline values in 10 CFR Part 100.

The NRC staff's review of this safety issue has been incorporated in the NRC Program for Resolution of Generic Issues as Generic Task A-36.

The objective of the task is to develop a revision to the Standard Review Plan (SRP) based on a reevaluation of current NRC requirements and procedures curretnly utilized at operating plants. If found to be necessary, the revision will provide criteria to further reduce the potential for heavy loads causing unacceptable damage to spent fuel in a storage pool or in the reactor core during refueling. The revised SRP will provide the basis for implementing additional requirements and procedures in existing plants where warranted and can be used in future reviews of new plants.

It is the NRC staff's view that continued operation du g our review of this generic issue presents no undue risk to the health and safety of the public. Operating facilities use a variety of design and administrative measures to minimize the potential for dropping a heavy object over the reactor core or over the spent fuel pool. These design and administrative measures have been effective since no heavy load handling accidents resulting in damaged fuel have occurred in over 300 reactor years of U.S. operating experience. Additionally, for facilities that have requested increases in spent fuel pool storage capacity, the NRC has prohibited the movement of loads over fuel assemblies in the spent fuel pool that weigh more than the equivalent weight of one fuel assembly. Also for those plants where a review of cask drop or the crane handling system is not complete, movement of shielded casks over or near spent fuel has been prohibited.

Concurrent with our review, licensees have reviewed their current procedures for the movement of heavy loads over spent fuel to assure that the potential for a handling accident that could result in damage of spent fuel is minimized while our generic evaluation proceeds. The majority of the licensees' submittals of their reviews have been received and are under review.

Generic Task A-36 is expected to be completed in early 1979. The Task will result in the development of generic criteria, however, implementation of these criteria will be highly dependent on plant design characteristics and the specific procedures in effect at each particular plant.

SEISMIC DESIGN CRITERIA
(GENERIC TASK A-40)

NRC Regulations require that nuclear power plant structures, systems and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes. Detailed requirements and guidance regarding the seismic design of nuclear plants is provided in the NRC regulations and in Regulatory Guides issued by the Commission. However, there are a number of plants with construction permits and operating licenses issued before the NRC's current regulations and regulatory guidance were in place. For this reason, rereviews of the seismic design of various plants are being undertaken (principally as part of the Commission's Systematic Evaluation Program) to assure that these plants do not present an un- risk to the public.

The NRC staff is conducting Generic Task A-40, as part of the NRC Program for Resolution of Generic Issues. Task A-40 is, in effect, a compendium of short-term efforts to support the reevaluation of the seismic design of operating reactors.

The objective of Task A-40 is, in part, to investigate selected areas of the seismic design sequence to determine their conservatism for all types of sites, to investigate alternate approaches to parts of the design sequence, and to quantify the overall conservatism of the design sequence. In this manner this program will aid the NRC staff in performing its reviews of the seismic design of operating reactors.

Generic Task A-40 is separated into ten separate subtasks. The majority of the subtasks are scheduled for completion in September 1979. However, three of the subtasks related to developing state-of-the-art methodology to better define earthquake ground motion near earthquake sources are longer term efforts. These three subtasks are scheduled for completion in 1981.

PIPE CRACKS AT BOILING WATER REACTORS
(GENERIC TASK A-42)

Pipe cracking has occurred in the heat affected zones of welds in primary system piping in boiling water reactors since the mid-1960s. These cracks have occurred mainly in Type 304 stainless steel that is being used in most operating BWRs. The major problem is recognized to be intergranular stress corrosion cracking (IGSCC) of austenitic stainless steel components that have been made susceptible to this failure mode by being "sensitized" either by post-weld heat treatment or by sensitization of a narrow heat affected zone near welds.

"Safe ends" (short transition pieces between vessel nozzles and the piping) that have been highly sensitized by furnace heat treatment while attached to vessels during fabrication were very early (late 1960s) found to be susceptible to IGSCC. Because they were susceptible to cracking, the Atomic Energy Commission took the position in 1969, that furnace sensitized safe ends should not be used on new applications. Most of the furnace sensitized safe ends in older plants have been removed or clad with a protective material, and there are only a few BWRs that still have furnace sensitized safe ends in use. Most of these, however, are in smaller diameter lines.

Earlier reported cracks (prior to 1975) occurred primarily in 4" diameter recirculation loop-by-pass lines and in 10" diameter core spray lines. More recently cracks were discovered in recirculation riser piping (12" - 14") in foreign plants. Cracking is most often detected during Inservice Inspection using ultrasonic testing techniques. Some piping cracks have been discovered as a result of primary coolant leaks.

Because of these occurrences of BWR primary system cracking, there has been a variety of actions undertaken by the NRC. These actions included:

- issuance of Regulatory Guide 1.44 on "Control of the Use of Sensitized Stainless Steel"
- issuance of Regulatory Guide 1.45 on "Reactor Coolant Boundry Leak Detection Systems"
- closely following the incidence of cracking in BWRs, including foreign experience
- encouraging replacement of furnace sensitized safe ends

402 356

- requiring augmented inservice inspection (additional more frequent ultrasonic examination) of "service sensitive" lines, i.e., those that have experienced cracking
- requiring upgrading of leak detection systems

Pipe cracking and furnace sensitized safe and cracking has been recently reported in larger (24" diameter) lines in a GE-designed BWR in Germany with over 10 years of service. Because the safe ends on that facility had been furnace sensitized during fabrication, IGSCC was suspected. As a result of concerns regarding these furnace sensitized safe ends, a safe end was removed in order to perform destructive examination. During laboratory examination of the removed safe end, including a small section of attached pipe, cracks were discovered at various locations in the safe end and in the weld heat affected zone of the pipe. The cracks in the pipe weld area were very shallow with the maximum depth less than about 5 mm (about 1/8"). Cracking in the furnace sensitized safe end was somewhat deeper. The German experience was the first known occurrence of IGSCC in pipes as large as 24" in diameter.

In June 1978, a through-wall crack was discovered in an Inconel recirculation riser safe end (10" diameter) at the Duane Arnold facility. The crack has been attributed to IGSCC although the material in this instance is different from the Type 304 stainless steel that has been historically found to crack. Subsequent ultrasonic examination discovered indications in six of the other seven safe ends. Following their removal, cracking was discovered in all eight safe ends. The cracking appeared to have originated in a tight crevice between the inside wall of the safe end and an internal thermal sleeve. Such crevices are known to enhance IGSCC. Differences in materials, geometry, stress levels and crevices appear to make the problem at Duane Arnold unique to a particular type of recirculation riser safe end (Type I). As a result of this event, ultrasonic examination of the other Type I safe ends in U.S. BWRs, i.e., at the Brunswick 1 and 2 facility, was conducted. No significant indications were found in

402 357

Unit 2 and one indication, was identified at Unit 1. Although this indication is relatively minor and is not "reportable" pursuant to the NRC Regulations, it is continuing to be evaluated. The ultrasonic indication which was found will be reevaluated at another plant shutdown scheduled for later in 1978.

In addition to discussions with General Electric (the reactor vendor) regarding recent pipe cracking experience, General Electric was asked to provide an in-depth report on the significance of recent events regarding current inspection, repair, and replacement programs. They were also asked to address any new safety concerns related to the occurrence of cracking in large main recirculation piping. Based on information presented by General Electric to date, and extensive staff evaluation, it was concluded that the recent occurrences do not constitute a basis for immediate concern about plant safety, nor require any new immediate actions by licensees.

The staff briefed the Commission on pipe cracking in BWRs on August 31, 1978, and on September 14, 1978, re-established an NRC Pipe Crack Study Group. The Study Group will specifically address the following issues:

- the significance of the cracks discovered in large diameter pipes relative to the conclusions and recommendations set forth in the referenced report and in its implementation document NUREG-0313,
- resolution of concerns raised over the ability to use ultrasonic techniques to detect cracks in austenitic stainless steel,
- the significance of the cracks found in large diameter sensitized safe ends, and any recommendations regarding the current NRC program for dealing with this matter,
- the potential for stress corrosion cracking in PWRs, and
- the significance of the safe end cracking at Duane Arnold relative to similar material and design aspects at other facilities.

The Study Group is scheduled to complete its evaluation and report in January 1979. In addition to the Study Group effort, the NRC has underway several generic technical review efforts regarding flaw detection which are aimed at improving piping inspection techniques and requirements. These generic efforts and any follow-on efforts resulting from the Study Group's evaluation will be incorporated into a new Category A generic task, Task A-42, "Pipe Cracks at Boiling Water Reactors."

402 358

CONTAINMENT EMERGENCY SUMP RELIABILITY
(GENERIC TASK A-43)

Following a postulated loss-of-coolant accident, i.e., a break in the reactor coolant system piping, water flowing from the break would be collected in the emergency sump at the low point in the containment. This water would be recirculated through the reactor system by the emergency core cooling pumps to maintain core cooling. This water would also be circulated through the containment spray system to remove heat and fission products from the containment. Loss of the ability to draw water from the emergency sump could disable the emergency core cooling and containment spray systems. The consequences of the resulting inability to cool the reactor core or the containment atmosphere could be melting of the core and/or breaking of the containment.

One postulated means of losing the ability to draw water from the emergency sump could be blockage by debris. A principal source of such debris could be the thermal insulation on the reactor coolant system piping. In the event of a piping break, the subsequent violent release of the high pressure water in the reactor coolant system could rip off the insulation in the area of the break. This debris could then be swept into the sump, potentially causing damage.

Currently, regulatory positions regarding sump design are presented in Regulatory Guide 1.8 , Sumps for Emergency Core Cooling and Containment Spray Systems," which addresses debris (insulation). The Regulatory Guide recommends, in addition to providing redundant separated sumps, that two protective screens be provided. A low approach velocity in the vicinity of the sump is required to allow insulation to settle out before reaching the sump screening; and it is required that the sump remain functional assuming that one-half of the screen surface area is blocked. The NRC staff believes that sump designs in accordance with this regulatory guide acceptably resolve this issue. Nonetheless, the NRC staff is continuing to study the behavior of insulation under pipe break conditions to gain a better understanding of how it might behave.

402 359

A second postulated means of losing the ability to draw water from the emergency sump could be abnormal conditions in the sump or at the pump inlet such as air entrainment, vortices, or excessive pressure drops. These conditions could result in pump cavitation, reduced flow and possible damage to the pumps.

Currently, regulatory positions regarding sump testing are contained in Regulatory Guide 1.79, "Pre-Operational Testing of Emergen Cooling Systems for Pressurized Water Reactors," which addresses the g of the recirculation function. Both in-plant and scale model tests have been performed to demonstrate that circulation through the sump can be reliably accomplished. The NRC staff believes that sumps tested in accordance with this Regulatory Guide acceptably resolve this issue. As supplemental guidance, the staff, through a contractor, is studying whether further guidance for the design and review of emergency sumps to assure adequate hydraulic design can be developed.

The NRC staff initially planned to study the issue of containment emergency sump blockage from insulation as part of Generic Task C-3, "Insulation Usage Within Containment." In addition, initial plans were to study the vortex formation issue as part of Generic Task B-18, "Vortex Suppression Requirements for Containments." However, containment emergency sump operability is fundamental to the successful operation of both the emergency core cooling system (needed to cool the core) and the containment spray system (needed to assure containment integrity) following a loss-of-coolant accident. For this reason, these portions of Tasks C-3 and B-18 have been combined and elevated to Category A as Generic Task A-43 under the more general title of "Containment Emergency Sump Reliability." Because this action has only recently been taken, a Task Action Plan and schedule for this task have not yet been developed.

STATION BLACKOUT
(GENERIC TASK A-44)

Electrical ›wer for safety systems at nuclear power plants is supplied by two redundant and independent divisions. The systems used to remove decay heat to cool the reactor core following a reactor shutdown are included among the safety systems that must meet these requirements. Each electrical divison for safety systems includes an offsite alternating current (a. c.) power connection, a standby emergency diesel generator a. c. power supply, and direct current (d. c.) sources.

The issue of station blackout was originally included as Generic Task B-57 in the NRC Program for Resolution of Generic Issues. The task involves a study of whether or not nuclear power plants should be designed to accommodate a complete loss of all a. c. power, i.e., a loss of offsite a. c. sources and both onsite emergency diesel generator sources. Loss of all a. c. for an extended period of time in pressurized water reactors accompanied by loss of the auxiliary feedwater pumps (usually one of two redundant pumps is a steam turbine driven pump that is not dependent on a. c. power for actuation or operation) could result in an inability to cool the reactor core with potentially serious consequences. If the auxiliary feedwater pumps are dependent on a. c. power to function, then a loss of all a. c. power for an extended period could of itself result in an inability to cool the reactor core. Although this is a low probability event sequence, it could be a significant contributor to risk.

Current NRC safety requirements require as a minimum that diverse power drives be provided for the redundant auxiliary feedwater pumps. As noted above, this is normally accomplished by utilizing an a. c. powered electr[illegible] motor driven pump and a redundant steam turbine driven pump. One [illegible] design adequacy of plants licensed prior to adoption of the [illegible] irements.

An initial survey of operating plants has been completed which indicates that all operating pressurized water reactors have either steam turbine driven or diesel driven auxiliary feedwater pumps (neither of which are dependent on a. c. power). This assures at least that some capability exists for accommodating an extended loss of all a. c. power. Further review of older plants in this regard will be conducted as part of the NRC's Systematic Evaluation Program (see earlier discussion in this Chapter on page [illegible]).

Further study regarding this issue will include determining if requirements beyond diverse power drives for the auxiliary feedwater pumps are needed. Such requirements might include specific time requirements for which the plant must be capable of accommodating a station blackout.

This safety issue was previously included in the NRC Program for the Resolu⁺ⁱ ⁿ of Generic Issues as Generic Task B-57, but has recently been elevated to Ca gory A as Generic Task A-44. Because this action has onl, recently been taken, a Task Action Plan and schedule for this task have not yet been developed.

402 362