UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
JERSEY CENTRAL POWER & LIGHT COMPANY	)	
	)	Docket No. 50-219
(Oyster Creek Nuclear Power Plant,	)	
Unit No. 1)	)	

AFFIDAVIT OF WALTER A. PAULSON

STATE OF MARYLAND)
COUNTY OF MONTGOMERY) SS

I, Walter A. Paulson, do hereby swear and depose as follows:

1. I am the Nuclear Regulatory Commission project manager for the Oyster Creek Nuclear Generating Station.

2. The attached information concerning unresolved generic safety issues applicable to the Oyster Creek plant was prepared by me or under my supervision.

3. The information provided in the attached document is true and correct to the best of my knowledge and belief.

Walter A. Paulson

Walter A. Paulson
Operating Reactor Branch No. 5
Office of Nuclear Reactor Regulation

Subscribed and sworn to before me
this 28th day of October, 1980

Elizabeth Ann Tyson
Notary Public

My Commission expires: July 1, 1982

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UNRESOLVED GENERIC SAFETY ISSUES (CATEGORY A)
AND OTHER ISSUES LISTED IN NUREG-0510 (CATEGORY A OR
CATEGORY B) WHICH MAY HAVE AN IMPACT ON THE
SAFE OPERATION OF THE OYSTER CREEK
NUCLEAR GENERATING STATION

Introduction

The NRC staff continuously evaluates the safety requirements used in its reviews against new information as it becomes available. Information related to the safety of nuclear power plants comes from a variety of sources including experience from operating reactors, research results, NRC staff and Advisory Committee on Reactor Safeguards safety reviews, and vendor, architect/engineer and utility design reviews. Each time a new concern or safety issue is identified from one or more of these sources, the need for immediate action to assure safe operation is assessed. This assessment includes consideration of the generic implications of the issue.

In some cases, immediate action is taken to assure safety, e.g., the derating of boiling water reactors as a result of the channel box wear problems in 1975. In other cases, interim measures, such as modifications to operating procedures, may be sufficient to allow further study of the issue prior to making licensing decisions. In most cases, however, the initial assessment indicates that immediate licensing actions or changes in licensing criteria are not necessary. In any event, further study may be deemed appropriate to make judgements as to whether existing NRC staff requirements should be modified to address the issue for new plants or if backfitting is appropriate for the long-term operation of plants already under construction or in operation.

Unresolved Generic Safety Issues

These issues are sometimes called "generic safety issues" because they are related to a particular class or type of nuclear facility rather than a specific plant. These issues have also been referred to as "unresolved safety issues." However, as discussed above, such issues are considered on a generic basis only after the staff has made an initial determination that the safety significance of the issue does not prohibit continued operation or require licensing actions while the longer-term generic review is underway.

As a result of Congressional action on the Nuclear Regulatory Commission budget for Fiscal Year 1978, the Energy Reorganization Act of 1974 was amended (PL 95-209) on December 13, 1977 to include, among other things, a new Section 210 as follows:

"UNRESOLVED SAFETY ISSUES PLAN"

> "SEC. 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 of the Commission thereafter."

The Joint Explanatory Statement of the House-Senate Conference Committee for the FY 1978 Appropriations Bill (Bill S.1131) provided the following additional

information regarding the Committee's deliberations on this portion of the bill:

"SECTION 3 - UNRESOLVED SAFETY ISSUES"

> "The House amendment required development of a plan to resolve generic safety issues. The conferees agreed to a requirement that the plan be submitted to the Congress on or before January 1, 1978. The conferees also expressed the intent that this plan should identify and describe those safety issues, relating to nuclear power reactors, which are unresolved on the date of enactment. It should set forth: (1) Commission actions taken directly or indirectly to develop and implement corrective measures; (2) further actions planned concerning such measures; and (3) timetables and cost estimates of such actions. The Commission should indicate the priority it has assigned to each issue, and the basis on which priorities have been assigned."

In response to the reporting requirements of the new Section 210, the Commission submitted to Congress on January 1, 1978, a report describing the NRC generic issues program (NUREG-0410).[1/] The NRC program was already in place when PL 95-209 was enacted and is of considerably broader scope than the "Unresolved Safety Issues Plan" required by Section 210. In the letter transmitting NUREG-0410 to the Congress dated December 30, 1977, the Commission indicated that "the progress reports, which are required by Section 210 to be included in future NRC annual reports, may be more useful to Congress if they focus on the specific Section 210 safety items."

It is the NRC's view that the intent of Section 210 was to assure that plans were developed and implemented on issues with potentially significant public

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1/ NUREG-0410, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," issued on January 1, 1978.

safety implications. In 1978, the NRC undertook a review of over 130 generic issues addressed in the NRC program to determine which issues fit this description and qualify as "Unresolved Safety Issues" for reporting to the Congress. The NRC review included the development of proposals by the NRC staff and review and final approval by the NRC Commissioners.

This review is described in a report, NUREG-0510, entitled "Identification of Unresolved Safety Issues Relating to Nuclear Power Plants - A Report to Congress" dated January 1979. The report provides the following definition of an "Unresolved Safety Issue:"

> "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 it affects."

Further the report indicates that in applying this definition, matters that pose "important questions concerning the adequacy of existing safety requirements" were judged to be those for which resolution is necessary to (1) compensate for a possible major reduction in the degree of protection of the public health and safety, or (2) provide a potentially significant decrease in the risk to the public health and safety. Quite simply, an "Unresolved Safety Issue" is potentially significant from a public safety standpoint and its resolution is likely to result in NRC action on the affected plants.

All of the issues addressed in the NRC program were systematically evaluated against this definition as described in NUREG-0510. As a result, 17 "Unresolved Safety Issues" addressed by 22 tasks in the NRC program were

identified.[2/] The issues are listed below. Progress on these issues was discussed in the 1979 NRC Annual Report. The number(s) of the generic task(s) (e.g., A-1) in the NRC program addressing each issue is indicated in parentheses following the title.

"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. Systems Interaction 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)

[2/] No additional "Unresolved Safety Issues" have been approved by the Commission for reporting to Congress. The 1979 Annual Report at page 66 includes a commitment to provide a special report to the Congress in July 1980 identifying new Unresolved Safety Issues following a systematic review of all candidate issues from the Three Mile Island investigations and other sources. A systematic review of candidate issues has been performed by the Office of Nuclear Reactor Regulation and the results of that review are pending before the Commission. SECY-80-325 (July 9, 1980).

13. Control of Heavy Loads Near Spent Fuel - (A-36)
14. Seismic Design Criteria - (A-40)
15. Pipe Cracks at Boiling Water Reactors - (A-42)
16. Containment Emergency Sump Reliability - (A-43)
17. Station Blackout - (A-44)

Six of the 22 tasks identified with the above 17 "Unresolved Safety Issues" (A-2, A-3, A-4, A-5, A-12 and A-26) are peculiar to pressurized water reactors and are therefore not applicable to the Oyster Creek Nuclear Generating Station. An additional Task, A-8, relates to the Mark II containment and is also not applicable to the Oyster Creek facility, whic uses a Mark I containment design.

Seven Additional Items

The staff also considered all issues (in either Category A or Category B of NUREG-0510) other than "unresolved safety issues," which were categorized as either "Potential High Risk Items" or "Potential Low Risk Items", in NUREG-0510 (Appendix C). This led to the identification of seven additional issues (A-29, B-30, B-34, B-55, B-63, A-15 and A-30) applicable to BWRs which were not treated as unresolved safety issues, but are identified as issues which may have an impact on the facilities and which, if left unresolved, could present potentially serious safety or environmental concerns.

These seven additional tasks are itemized below:

1. Design Features to Control Sabotage - (A-29)
2. Isolation of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary - (B-63)

3. Design Basis Floods and Probability - (B-30)
4. Occupational Radiation Exposure Reduction - (B-34)
5. Improved Reliability of Target Rock Safety Relief Valves - (B-55)[3/]
6. DC Power Supplies - (A-30)
7. Decontamination - (A-15)

Summary

The above 17 "Unresolved Safety Issues" and six additional items constitute all of the unresolved matters of potential safety significance for the Oyster Creek facility.

We have reviewed each of the above identified issues as they relate to the Oyster Creek facility. Discussions of each of the applicable issues are provided below in Section 1 - Unresolved Safety Issues and Section 2 - Six Additional Items. For each issue, there is provided a brief task description and a discussion setting forth the Staff's bases for its conclusion that the Oyster Creek Nuclear Generating Station can continue to be operated prior to the ultimate resolution of these generic issues without endangering the health and safety of the public.

[3/] Task B-55, "Improved Reliability of Target Rock Safety Relief Valves" is not applicable to the Oyster Creek facility which utilizes Electromatic safety relief valves.

SECTION I

UNRESOLVED SAFETY ISSUES

TASK A-1 WATER HAMMER

Task Description

Water hammer events are intense pressure pulses in fluid systems caused by any one of a number of mechanisms and system conditions such as rapid condensation of steam pockets, steam-driven slugs of water, pump startup with partially empty lines, and rapid valve motion. Since 1971, over 200 incidents involving water hammer in pressurized and boiling water reactors have been reported.

The water hammers (or steam hammers) have involved steam generator feedrings and piping, the residual heat removal system, emergency core cooling systems, and containment spray, service water, feedwater and steam lines.

Most of the damage reported has been relatively minor, involving pipe hangers and restraints; however, several water hammer incidents have resulted in piping and valve damage.

The most serious water hammer events have occurred in the steam generator feed rings of pressurized water reactors. In no case has any water hammer incident resulted in the release of radioactive material.

Under Generic Task A-1, 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 review. A technical report, NUREG-0582 "Water Hammer in Nuclear Power Plants," (July 1979) providing the results of an NRC staff review of water hammer events in nuclear power plants and stating current staff licensing positions, completes a major subtask of Generic Task A-1.

Basis for Continued Plant Operation Pending Completion of Task A-1

Although water hammer can occur in any LWR and approximately 118 actual and probable events have been reported in BWRs as of September 1979 (of which four have occurred at Oyster Creek), none have caused major pipe failures in a BWR and none have resulted in the off-site release of radioactivity. One water hammer event which occurred during power operation at Oyster Creek on December 11, 1971 resulted in failure of an isolation condenser vent line. Investigation of the event showed that the pipe failure could have been prevented if the pipe hanger called for in the plant design had been installed. Subsequently, a hanger was installed on this line. Adequate protection from potential loss-of-coolant accidents due to pipe failures (such as might be initiated by a water hammer event) is provided at Oyster Creek by the emergency core cooling systems, which provide protection in accordance with the ECCS performance requirements of 10 CFR 50.46 and 10 CFR 50, Appendix K over the entire spectrum of postulated pipe break sizes. Since operating experience at Oyster Creek and other BWRs has shown the probability of a water hammer event causing pipe failure is low in a BWR and since the consequences of

postulated water hammer induced accidents would be adequately limited by the currently installed ECCS at Oyster Creek, continued operation of the Oyster Creek facility pending completion of this task does not present an undue risk to the health and safety of the public.

TASKS A-6, A-7 AND A-39 - BWR MARK I SUPPRESSION CONTAINMENTS

Boiling water reactor pressure suppression containments designed by the General Electric Company utilize a large mass of water as the principal heat sink by which to condense the steam and absorb the energy that might be released in the event of a loss-of-coolant accident (LOCA). During the conduct of a large scale testing program for an advanced design pressure-suppression containment system (Mark III) for BWRs, new suppression pool hydrodynamic loads associated with a postulated LOCA were identified which had not been explicitly included in the original design of the Mark I containment systems, including the Oyster Creek system. These additional loads result from dynamic effects of drywell air and steam being rapidly forced into the suppression pool (torus) during a postulated LOCA event. In addition, experience at operating plants has indicated that the dynamic effects of safety-relief valve (SRV) discharges to the suppression pool could be substantial and should be reconsidered.

The affected utilities formed Owner's Groups and established both short term and long term programs for the resolution of the pool dynamics problems. The programs include a number of comprehensive experimental and analytical programs to establish pool dynamic loads, load combinations, and design criteria. The NRC staff identified and initiated a number of generic tasks to review and evaluate the results of the industry programs and to develop criteria for licensing actions on individual plants using the Mark I containments.

A. Task A-6, Mark I Short Term Program

Task Description

The objectives of Task A-6, "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 it would maintain its integrity and functional capability when subjected to the most probable hydrodynamic loads induced by a postulated design basis LOCA, and (2) to verify that the 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. Based on our review, we determined that (1) the magnitude and character of each of the hydro-dynamic loads resulting from a postulated LOCA have been adequately defined for use in the Short-Term Program structural assessment of the Mark I Containment System, (2) for the most probable loads induced by a postulated LOCA, a safety factor to failure of at least two exists for the weakest structural or mechanical component in the containment system for the Oyster Creek facility and (3) that based on (1) and (2), the

Oyster Creek Mark I Containment System would maintain its integrity and functional capability in the unlikely event of a design basis LOCA. We concluded that the objectives of the short-term program had been satisfied and documented the basis for this conclusion in NUREG-0408, "Mark I Containment Short-Term Program Safety Evaluation Report", December 1977. Subsequently, the NRC granted the operating Mark I facilities exemptions relating to the structural factor of safety requirements of 10 CFR 50.55(a) for an interim period of about two years, while the more methodical long-term program was being conducted. Thus, Task A-6 has been completed.

Basis for Continued Plant Operation Pending Completion of Task A-6

As discussed above, Task A-6 has been completed for Oyster Creek.

B. Task A-7 - Mark I Long-Term Program

Task Description

The objectives of Task A-7, "Mark I Long-Term Programs" are: (1) to establish design basis (conservative) loads that are appropriate for the anticipated life (40 years) of each Mark I BWR facility, and (2) to make whatever plant changes may be required to restore the original intended design safety margins for each Mark I containment system. The NRC staff's evaluation of the generic load definition and structural assessment techniques that have been proposed by the Mark I Owners Group as part of the long-term program has been completed and is documented in NUREG-0661, "Safety Evaluation Report Mark I Containment Long-Term Program," July 1980. The requirements which have resulted from the NRC staff evaluation will be used by each BWR/Mark I licensee to perform plant unique analyses. These analyses will serve to identify those plant modifications that are needed to restore the margins of safety in the containment design. Although the Oyster Creek licensee has been designing and installing plant modifications throughout the course of the long-term program, the staff has requested that the Commission approve the issuance of Orders to all Mark I licensees to confirm their completion schedules (SECY-80-359 and SECY-80-359A). For Oyster Creek these modifications are scheduled to be completed during the Fall 1981 refueling outage.

Basis for Continued Plant Operation Pending Completion of Task A-7

As documented in NUREG-0408, "Mark I Containment Short-Term Program Safety Evaluation Report," December 1977, based upon our review of the generic "Short-Term Program Final Report" and Addenda submitted by the Mark I Owner's Group and the plant-unique analysis reports submitted by Jersey Central Power & Power Company for the Oyster Creek facility, we have concluded that Oyster Creek can continue to operate safely, without undue risk to the health and safety of the public, while a methodical, comprehensive Long-Term Program (LTP) is conducted and the necessary modifications are completed which, for Oyster Creek, are scheduled for the Fall 1981 refueling outage.

This conclusion has been made based on our determination: (1) that the magnitude and character of each of the hydrodynamic loads resulting from a postulated LOCA have been adequately defined for use in the Short-Term Program (STP) structural assessment of the Mark I Containment System, (2) that, for the "most probable" loads induced by a postulated LOCA, a safety factor to failure of at least two exists for the weakest structural or mechanical component in the containment system for the Oyster Creek facility, and (3) that, based on (1) and (2) above, the Oyster Creek Mark I Containment System would maintain its integrity and functional capability in the unlikely event of a design basis LOCA.

C. Task A-39 - Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits for BWR Containments

Task Description

The objective of Task A-39, "Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits for BWR Containments" is to review and evaluate the results of industry experimental and analytical programs to establish and justify the safety relief valve-related pool dynamics loads for BWR pressure suppression designs including the Mark I containments. The short-term program portion of this task is addressed in NUREG-0408, "Mark I Containment Short-Term Program Safety Evaluation Report." The NRC staff's evaluation of safety relief valve discharge loads for the Mark I containment is described in NUREG-0661. Modifications are to be completed on the same schedule as that discussed above for Task A-7 modifications.

Basis for Continued Plant Operation Pending Completion of Task A-39

With regard to safety relief valve loads, the justification for continued operation is based on our evaluation of operating experiences and the plant capability to tolerate SRV loads. SRV operating experience has shown that in all but a few instances, SRV discharges have performed satisfactorily without any evidence of damage either due to the hydrodynamic loads or pool temperature effect. In those isolated cases where localized damage has been encountered, the damage did not result in a loss of the containment function, or release of radioactivity. In those cases, repairs were made and additional margin was included in the structures. Oyster Creek will be required to demonstrate the capability to meet the SRV loads criteria and pool temperature limit which have been established by this task and presented in NUREG-0661.

In particular, Oyster Creek has installed steam quencher devices on the SRV discharge. In-plant tests were also performed during 1977 and 1978. Results of the tests show that the SRV loads are reduced by a factor from two to four. The staff concludes that Oyster Creek has the capability to tolerate SRV loads.

We conclude that operation of Oyster Creek can continue prior to resolution of this generic issue without undue risk to the health and safety of the public.

TASK A-9 ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

Task Description

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 reactor (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.

In 1978, the NRC staff proposed a combination of preventative and mitigative means of providing protection from ATWS events (Volume 3 of NUREG-0460). Volume 4 of NUREG-0460, issued for comment in March 1980 presented the results of the NRC staff review of industry responses to the alternatives proposed in Volume 3. The NRC staff has received the industry and ACRS comments and has prepared a recommendation for rulemaking for Commission consideration. As an interim measure, all BWRs have been required to install a recirculation pump trip to mitigate the consequences of an ATWS.

Basis for Continued Plant Operation Pending Completion of Task A-9

The recirculation pump trip system in boiling water reactors helps limit the consequences of an ATWS event. The Oyster Creek facility incorporates this feature. In addition, the licensee has implemented a procedure for recognizing a failure to scram event and instructing the operator regarding actions to be taken to mitigate the effects of such an event. The licensee has stated that the operators have been trained to cope with a failure-to-scram event.

Operator training and action as described, in conjunction with the automatic recirculation pump trip, significantly improves the capability of the facility to withstand a range of ATWS events, such that continued operation of this facility presents no undue risk to the health and safety of the public while this matter is under review.

TASK A-10 BWR NOZZLE CRACKING

Task Description

A) Feedwater Nozzle

Over the past several years, inspections at 22 of the 23 boiling water reactor (BWR) plants in the United States that have feedwater nozzle/sparger

systems have disclosed some degree of cracking in the feedwater nozzles of the reactor vessels at 18 plants. (One plant has not accumulated significant operating time as of the date of this report and has not been inspected.) Similar cracking has occurred in BWR control rod drive return line nozzles. Both issues are considered by the staff to be satisfactorly resolved, with the exception of the development of improved nondestructive examination (NDE) techniques.

Most BWR pressure vessels have four feedwater nozzles. Several vessels have six such nozzles and one vessel has only one nozzle. Three older plants do not have feedwater nozzles per se. Nozzle diameter is 10 to 12 inches, depending on plant design.

The feedwater is distributed through spargers that deliver the flow evenly to assure proper subcooling and help maintain proper core power distribution. An essential part of the sparger is the thermal sleeve, which projects into the nozzle bore and is intended to prevent the impingement of cold feedwater on the hot nozzle surface. This surface is usually heated to essentially reactor water temperature by the returning water from the steam separators and steam dryers. However, bypass leakage past the thermal sleeves allowed relatively cold feedwater to impinge on the hot nozzles. The feedwater, when heated during power operation by extracting steam from the main turbine, is typically about 100 F to 200 F colder (depending on reactor design) than the reactor water. When the feedwater heaters are not in service, as during startups and shutdowns, the differential could be equal to or greater than 400 F. The bypass leakage past a loose thermal sleeve caused a fluctuation (at times severe) in the metal temperature of the feedwater nozzle and resulted in metal fatigue and crack initiation. The cracks were then driven deeper by the larger temperature and pressure cycles associated with startups, shutdowns, and certain operational transients.

The older sparger design has been replaced by other designs. The tight fit, forged-tee design is used predominantly today as an interim measure until the installation of the modified triple-sleeve spargers or other acceptable designs. Several plants have welded thermal sleeve design. The staff believes the new designs provide a substantial and acceptable improvement over previous designs and should resolve the problems. Our evaluation and conclusions are documented in NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking".

Nozzle cracking is potentially serious for the following reasons:

(1) Excessive crack growth could lead to reduction of pressure vessel safety margins.

(2) The design safety margin could also be reduced by excessive removal of nozzle reinforcement metal when cracks are removed by grinding.

(3) The exposure to radiation of the personnel performing inspection and repair tasks can be considerable.

The substitution of tight-fitting, interference-fit spargers and increased licensee attention to proper feedwater control has significantly reduced the incidence of cracking in recent years.

The NRC performed an independent review of the nozzle cracking problem and evaluated the General Electric Company (GE) test data for confirmation of the causes of cracking and the resultant solutions. The staff concluded that the GE triple-sleeve sparger modification, when combined with removal of stainless steel cladding, feedwater system modifications when necessary, and changes to operating procedures, provides a substantial and acceptable improvement over previous designs. However, the staff recognizes that the GE design is not the only effective sparger modification. Another design has already been approved for use at two operating reactors. In any case, a reactor vessel modified with an improved sparger and other physical and procedural changes being incorporated as necessary, should be able to operate for an extended period of time between in-vessel nozzle surface examination.

Non-destructive examination will require continuing effort of the NRC staff and various industry groups. The industry efforts, which the staff will review, are directed toward the development of ultrasonic testing procedures that will find and characterize tight fatigue cracks in the complex geometries and long examination metal paths of BWR feedwater nozzles. Until the NRC staff is assured that such techniques are capable of reliably detecting flaws before they violate ASME Code flaw size and reinforcement limits, we will require in-vessel dye-penetrant surface examination (PTs). Upon completion of the industry studies mentioned above, the staff will issue further guidance on inservice inspection (ISI) requirements.

All repaired feedwater nozzles to date have met the requirements and limits of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. No additional action was necessary since only relatively small amounts of base metal have been removed by repair operations. The removal of cladding, as a means of minimizing crack initiation, has not altered the safety margins because the clad thickness is not considered in ASME Code reinforement requirements.

B) Control Rod Drive Return Line Nozzles

Twenty-two of the 23 operating BWR reactor vessels in the United States with feedwater nozzles/sparger systems also have control rod drive return lines (CRDRL) nozzles. Each vessel has one such nozzle, typically 3-4 inches in diameter, and generally located 68-100 inches above the top of the active fuel.

The control rod drive (CRD) system provides water to: (1) maintain rod scram accumulators in a charged condition at greater than reactor pressure; (2) drive the rods into or out of the core; and (3) cool the rod drive mechanisms continuously. The CRDRL was designed to provide a reactor pressure reference to the CRD system and to return to the reactor vessel exhaust water from CRD movement and water in excess of system requirements.

The cause of crack initiation of CRDRL nozzles is a thermal fatigue mechanism similar to that seen in feedwater nozzles. High-frequency thermal cycling occurs during normal operation as a result of turbulent mixing of hot water in the vessel with the low temperature (50 to 100 F) water entering through the CRDRL. Low-cycle fatigue crack propagation results from startup/ shutdown thermal and pressure cycles and from flow changes during scrams. In those plants that have a thermal sleeve in the CRDRL nozzle, bypass leakage flow is minimal because the pressure drop is much smaller than in feedwater nozzles, which have a thermal sleeve and sparger. In the CRDRL nozzle, unlike the feedwater nozzle, there is a continuous large top-to-bottom thermal gradient, which aggravates the cracking.

Also unlike the feedwater nozzle, cracks have been observed on the vessel wall directly beneath the CRDRL nozzle in an area extending downward 6-8 inches from the nozzle bend radius. The cracks on the vessel wall are mainly circumferential. They are believed to result from high-cycle thermal stresses related to stratified flow of cold water along the bottom of the nozzle and down the vessel wall as it mixes with the downflow of reactor water.

A GE study of the CRDRL nozzle cracking problem resulted in a series of recommendations to licensees. The NRC staff has reviewed each GE recommendation and has determined that (1) valving out of the return line is acceptable only as an interim measure; (2) rerouting of the return line to another system which connects to the reactor vessel is preferable, and (3) only certain BWR classes may implement the GE recommendation, to cut and cap the line and nozzle without rerouting, and then only after specific testing has been completed.

A detailed discussion regarding the CRD problem, the proposed solutions, and the NRC staff's review and conclusions is documented in NUREG-0619.

NUREG-0619, which was issued for public comment in May 1980 will be issued in final form in November 1980.

Basis for Continued Plant Operation Pending Completion of Task A-10

During the time period required to develop the long-term solutions under this task, interim measures have been taken. Specifically, the staff is requiring inservice inspection using liquid penetrant examinations at Oyster Creek in accordance with the procedures and acceptance criteria set forth in detail in NUREG-0312, "Interim Technical Report on BWR Feedwater and Control Rod Drive Return Line Nozzle Cracking," July 1977. Licensees are also utilizing ultrasonic inspection techniques in an effort to develop effective techniques that will allow early detection of subsurface flaws. Enhancement of ultrasonic testing techniques will substantially reduce personnel exposures. The scheduling and extent of inspection is based upon conservative estimates of crack growth from fracture mechanics analyses assuming undetected flaws. Scheduling is thus dependent upon the reactor's record of past repair (grindouts, clad removal, etc.), operating history

(number of startup/ shutdown cycles since last dye-penetrant inspection), and licensee actions to minimize crack initiation by procedural or mechanical change.

The staff has been actively involved in reviewing and approving the results of nozzle inspections and remedial actions proposed by licensees to assure continued safe operation. To date the extent of nozzle cracking at operating plants has been limited to depths which can be removed by grinding without exceeding ASME code limits for nozzle reinforcement.

In addition the staff has suggested that measures be taken at affected operating plants and by applicants for plants in the operating license review stage prior to operation, to minimize the occurrence of conditions conducive to crack initiation and growth. These measures include monitoring feedwater temperatures and flow, minimizing the duration of cold feedwater injection, avoiding inadvertent or unnecessary HPCI injection, avoiding the unnecessary introduction of cold water from the reactor water cleanup system, and eliminating flow through the control rod drive return line (after assuring proper system operation in an isolated mode). Although avoidance of cracking of the pressure vessel nozzles is important to safety, NRC staff analyses indicate that cracking that has penetrated the vessel cladding will grow at a slow enough rate such that the cracking does not pose a critical safety concern today that warrants immediate action. Rather, the staff believes that sufficient time is available, due to the conservative design of the reactor pressure vessel, to permit continued operation of the affected facilities while studies of these events continues.

Oyster Creek has installed, and the staff has approved a modified feedwater thermal sleeve/sparger combination and has implemented the specific inservice inspection requirements outlined in NUREG-0619. These requirements are based on a conservative determination by the NRC staff that the number of startup and shutdown thermal cycles prior to crack initiation using the modified design at Oyster Creek is sufficiently high so that nozzle cracking is not likely to occur. Inspection of the control rod drive return line nozzle at Oyster Creek indicated that the thermal sleeve was securely welded in place and that neither leakage nor cracking had occurred. The NRC staff has concluded that Oyster Creek can operate with the present design, provided that the dye penetrant tests specified in NUREG-0619 is performed as will be required by the staff.

Based on the interim measures being taken at Oyster Creek and the design margins available in the reactor pressure vessel, we have concluded that operation of Oyster Creek does not present an undue risk to the health and safety of the public.

TASK A-11 REACTOR VESSEL MATERIALS TOUGHNESS

Task Description

Prevention of reactor vessel failure depends primarily on maintaining the reactor vessel material fracture toughness at levels that will resist brittle fracture during plant operation. For the length of service and operating conditions typical of current operating plants, reactor vessel fracture toughness properties provide adequate margins of safety against vessel failure; however, as plants accumulate more and more service time, neutron irradiation reduces the material fracture toughness and initial safety margins.

To ensure adequate toughness margins for operating plants, the staff is developing (1) improved safety criteria for low toughness materials, (2) definition and identification of critical materials in reactor vessels, and (3) a program to monitor and evaluate operational materials surveillance program results relevant to establishing a suitable generic toughness criterion. For those facilities not yet licensed for operation, current licensing criteria are adequate to ensure suitable safety margins throughout design life with the materials currently employed for reactor vessel fabrication. However, the need exists to reconsider these current criteria in light of new methods that may be developed in the evaluation of low toughness materials and to appropriately augment or refine these present criteria to include these new aspects and maintain NRC licensing consistency.

As discussed above, the safety issue addressed by this task is the reduction of reactor vessel material fracture toughness as a result of neutron irradiation. The operational temperature range includes the transition temperature region, where material toughness increases significantly with increasing temperatures, and the upper shelf temperature region, where material toughness reaches a relatively constant maximum value. The task will develop licensing criteria to ensure that adequate margins of safety, relative to flaw-induced fracture, are maintained during normal operating and postulated accident conditions for reactor vessels containing beltline (that part of the reactor vessel directly opposite the core) material with reduced toughness after prolonged irradiation.

Basis for Continued Plant Operation Pending Completion of Task A-11

Pending completion of this task, the safety margins required by Appendix G to 10 CFR Part 50 for operation of Oyster Creek in the transition temperature region can be maintained during normal operation by appropriate shifts of the operating pressure-temperature limits as dictated by the material surveillance program results and discussed in Regulatory Guide 1.99. Initial analyses submitted by some NSSS vendors and our preliminary review indicate that adequate toughness margins can also be maintained in the transition region for postulated accident conditions for up to approximately 20 years of neutron irradiation, or significantly beyond completion of this task.

Should the results of this task indicate that in the future adequate margins of safety for the Oyster Creek reactor vessel cannot be demonstrated for both normal operation and postulated accident conditions, one or more of the following alternative measures can be taken.

(1) Reactor vessel annealing to regain material toughness in the beltline region.

(2) Increased beltline inspections using improved in-service inspection (ISI) techniques, as they become available with demonstrated required reliability, leading to a justified decrease in postulated flaw size.

(3) Modifications to the vessel internals or core design to modify the neutron flux and reduce subsequent material degradation.

(4) System modifications to limit the severity of loading (stress levels) of the reactor vessel during postulated emergency or accident conditions.

In summary, the staff considers that in the interim period the safety margins are adequate to ensure the safety of the Oyster Creek reactor vessel. Accordingly, we conclude that while the task is being performed, continued operation of Oyster Creek can proceed without endangering the health and safety of the public.

TASK A-17 SYSTEMS INTERACTIONS IN NUCLEAR POWER PLANTS

Task Description

This task addresses the development of a systematic process to review plant systems to determine their impact on other plant systems. The purpose of the task is to identify where the present design, analysis and review procedures may not acceptably account for potentially adverse systems interaction and to recommend the regulatory action that should be taken to rectify deficiencies. It is anticipated that this task will confirm that current licensing requirements and procedures acceptably control the potential for adverse systems interactions, even though some modifications for improvement in the review procedures and licensing requirements may be made.

Current licensing requirements are founded on the principle of defense-in-depth. Adherence to this principle results in requirements such as physical separation and independence of redundant safety systems, and protection against events such as high energy line ruptures, missiles, high winds, flooding, seismic events, fires, operator errors, and sabotage. These design provisions and licensing procedures provide adequate assurance against the potential for adverse system interactions. The quality assurance program which is followed during the design, construction, and operational phases for each plant is expected to provide added assurance against the potential for adverse systems interactions.

In November 1974, the Advisory Committee on Reactor Safeguards requested that the NRC staff give attention to the evaluation of safety systems from a multi-disciplinary point of view, in order 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 sufficiently integrated in their design and analysis activities to enable them to identify adverse interactions between and among systems. Such adverse events might occur, for example, because designers did not assure that redundancy and independence of safety systems were provided under all conditions of operation required, which might happen if the functional teams were not adequately coordinated.

In mid-1977, Task A-17 was initiated to confirm that present review procedures and safety criteria provide an acceptable level of redundancy and independence for systems required for safety by evaluating the potential for undesirable interactions between and among systems.

Basis for Continued Plant Operation Pending Completion of Task A-17

The NRC staff's current review procedures assign primary responsibility for review of various technical areas and safety systems to specific 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 confirm the adequacy of current review procedures.

Licensing procedures that are being followed for Oyster Creek under the Systematic Evaluation Program provide additional assurance against potentially adverse systems interactions. Experience to date has demonstrated that operating plants have been designed to provide reasonable assurance that adverse systems interactions will not occur.

Accordingly, since no specific safety issue is yet identified in this task, for the reasons stated above, we conclude that while this task is being performed, continued operation of Oyster Creek can proceed without endangering the health and safety of the public.

TASK A-24 ENVIRONMENTAL QUALIFICATION OF SAFETY RELATED ELECTRICAL EQUIPMENT

Task Description

In addition to 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. Certain of these postulated accidents could create severe environmental conditions inside the containment. The most serious of these accidents would be a high energy pipe break in the reactor coolant system piping or in a main steam line. In either case, the release of hot pressurized water and steam to the containment would create a high temperature environment (250 to 400°F) at high humidity (including steam) and pressure (as high as c. 50 psig). Additionally, some electrical equipment may 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.

In order to assure that electrical equipment in safety systems will perform its function under accident conditions, the NRC requires that such equipment -- principally equipment associated with the emergency core cooling system and containment isolation and cleanup systems -- be "environmentally qualified." 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 transmit 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 applies certain criteria for confirming the capability of electrical equipment important to safety 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, they have come to be defined in clearer detail over the years.

The process of clarifying the criteria has given rise to certain questions regarding: (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 be able to function (i.e., the "qualification of equipment" in these older plants), and (2) the adequacy of tests 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).

Basis for Continued Plant Operation Pending Completion of Task A-24

On February 8, 1979, licensees were requested by IE Bulletin 79-01 to examine safety related electrical equipment and ensure documentation of its qualification to function under postulated accident conditions. IE Bulletin 79-01B, sent to licensees on January 14, 1980, requested additional information as

well as that requested in Bulletin 79-01.[1/] The Commission's Memorandum and Order of May 23, 1980 (CLI 80-21) addressed safety related electrical equipment inside or outside containment, and requires that all such equipment in operating reactors be fully qualified by June 30, 1982. On August 29, 1980, Orders for Modification of License were issued to all operating reactor licensees requiring them to submit not later than November 1, 1980 information which fully and completely responds to the staff's request for information as specified in IE Bulletin 79-01B. In each case where equipment is identified as not being in compliance with qualification guidelines, the staff has required by November 1, 1980, either equipment replacement or corrective action (e.g., equipment relocation or modification) or justification for continued operation while outstanding concerns are being resolved.

In view of the immediacy of the interim resolution of this issue for all licensees, there is sufficient basis to allow plant operation pending ultimate resolution of the issue addressed by this task. The staff, therefore, concludes that while this task is being performed, continued operation of Oyster Creek can proceed without endangering the health and safety of the public.

TASK A-31 RESIDUAL HEAT REMOVAL SHUTDOWN REQUIREMENTS

Task Description

The safe shutdown of the 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 shut down, but system temperature and pressure are still at or near their normal operating values). Considerable emphasis has been placed on the hot standby condition of a power plant in the event of an accident or abnormal occurrence. A similar emphasis has been placed on long-term cooling, which is typically achieved by the residual heat removal (RHR) system. The RHR system can operate only when the reactor coolant pressure and temperatures have been reduced to substantially below their not 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 events sometimes require eventual cooldown and long-term cooling until the reactor coolant system is cold enough to perform inspection and repairs. For this reason, the ability to transfer heat from the reactor to the environment after a shutdown is an important safety function for both PWRs and BWRs. It is

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[1/] Bulletin 79-01B was not sent to licensees for plants such as Oyster Creek, under review as part of the staff's Systematic Evaluation Program. The information sought by Bulletin 79-01B was requested from these licensees by a series of letters and meetings during the months of February and March, 1980.

essential that a power plant be able to go from hot standby to cold shutdown conditions (when this is determined to be the safest course of action) under any normal or accident conditions.

This issue was included in the NRC Program for Resolution of Generic Issues as Generic Task A-31, "RHR Shutdown Requirements." In accordance 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 NRC's 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 of NRR. Accordingly, Generic Task A-31 has been completed and Standard Review Plan Section 5.4.7 has been modified to incorporate the approved changes.

In addition, 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.

Following the development of new safety requirements resulting from the TMI-2 accident, Regulatory Guide 1.139 was updated to consider those TMI-2 requirements pertinent to Residual Heat Removal. The ACRS agreed that the Regulatory Guide could be reissued for public comment. The Regulatory Guide is scheduled to be reissued for comment during calendar year 1980.

Basis for Continued Plant Operation Pending Completion of Task A-31

A detailed review of safe shutdown systems at Oyster Creek was conducted as part of the Systematic Evaluation Program (SEP) review. The review included specific system and equipment requirements for bringing the reactor to a cold shutdown condition and remaining in a cold shutdown condition. This review included a comparison with the provisions of Standard Review Plan (SRP) 5.4.7 and Branch Technical Position (BTP) 5-1. The NRC staff concluded that Oyster Creek is in essential compliance with SRP 5.4.7 and BTP 5-1. (Our current evaluation of the Oyster Creek Safe Shutdown Systems was forwarded to the licensee on September 25, 1980.)

TASK A-36 CONTROL OF HEAVY LOADS NEAR SPENT FUEL

Task Description

Overhead cranes are used to lift heavy objects, sometimes in the vicinity of spent fuel, in both PWRs and BWRs. If a heavy object, such as a spent fuel

shipping cask or shielding block, were to fall or tip onto spent fuel in the storage pool or in 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 in-plant 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. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" has been issued (July 1980) providing the staff's resolution of Task A-36. This report contains criteria that should be satisfied to assure the safe handling of heavy loads. It also contains various measures that should be implemented in the short term to provide an adequate level of protection while the longer term measures of the NUREG-0612 criteria are being implemented. Within the next 30 days, the staff plans to issue letters to all licensees requesting implementation of these short-term or interim measures within 90 days, and conformance to all of the NUREG-0612 criteria within 2 1/2 years.

Basis for Continued Plant Operation Pending Completion of Task A-36

It is the NRC staff's view that continued operation during 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 release of radioactivity have occurred in over 400 reactor years of United States operating experience. 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.

Concurrent with the NRC review, licensees (including Jersey Central Power & Light Company) have examined 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 the generic evaluation proceeds.

The further short term actions that will be implemented by approximately March 1, 1981 will include procedural improvements such as definition of safe load handling paths, crane operator training and crane operation precautions, crane inspection and testing, and procedures documenting proper methods for handling heavy loads. These short term actions will further reduce the likelihood of a load handling accident damaging irradiated fuel.

Based on the safe operational history of Oyster Creek, the interim actions taken by the licensee, and the additional short term actions required by the

staff, we find that continued operation pending final compliance with NUREG-0612 presents no undue risk to the health and safety of the public.

TASK A-40 SEISMIC DESIGN CRITERIA

Task Description

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. 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 these reasons, 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 undue 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 a compendium of short-term efforts to support the reevaluation of the seismic design of operating reactors, and to support licensing activity in general. The objective of the task is, in part, to investigate selected areas of the seismic design sequence to determine the conservatism for all types of sites, to investigate alternative approaches to part of the design sequence, and to estimate quantitatively the overall conservatism of the design sequence. In this manner the program will aid the NRC staff in performing its reviews of the seismic design of operating reactors.

The NRC Office of Nuclear Regulatory Research is also undertaking a related, but more comprehensive and long-term program to develop mathematical models to realistically predict the probability of radioactive releases from seismically induced events in nuclear power plants. This Seismic Safety Margin Research Program will utilize input from Task A-40 in a number of areas.

Generic Task A-40 is subdivided into two phases. Phase I includes a number of subtasks related to the response of structures, systems, and components to earthquakes. These subtasks include studies on: (1) quantifying conservatisms in seismic design, (2) electro-plastic seismic analysis methods, (3) site-specific response spectra, (4) nonlinear structural dynamic analysis procedures, and (5) soil structure interaction. These studies were performed under NRC sponsored contracts and all were completed by October 1979. Review of the results of these studies is underway. The results will support the seismic reevaluation of operating plants, particularly in the area of site-specific definition of seismic input.

Phase II of Task A-40 includes several subtasks related to numerical modeling of earthquake motion at the source, analysis of near source ground motion, and attentuation of high-frequency ground motion. Studies under these subtasks being conducted by NRC contractors are scheduled for completion by the end of 1980. Review and implementation of the results of these studies in terms of recommended revisions to the Standard Review Plan and Regulatory Guides are scheduled for March 1981.

Basis for Continued Plant Operation Pending Completion of Task A-40

Seismic reevaluation of the Oyster Creek facility is being conducted as part of the Systematic Evaluation Program (SEP). The development of site specific spectra at Oyster Creek was performed under the SEP program. The SEP is conducting a seismic review using the spectra specified in Regulatory Guide 1.60 with a peak ground acceleration of 0.2g. The SEP seismic review currently underway is scheduled to be completed by the summer 1981.

Based on the preliminary review that has been performed and the conservatism resulting from the use of the Regulatory Guide 1.60 spectra with 0.2g acceleration, we conclude that pending completion of the seismic reevaluation, the continued operation of the Oyster Creek facility does not present undue risk to the health and safety of the public.

TASK A-42 PIPE CRACKS IN BOILING WATER REACTORS

Task Description

Leaks and cracks in the heat-affected zones (HAZs) of welds that join austenitic stainless steel piping and associated components in BWRs have been observed since the mid-1960s. Prior to September 1974, all affected piping was Type 304 stainless steel with diameters of eight inches or less. All the cracks were attributed to intergranular stress corrosion cracking (IGSCC) due to the combination of high local stress, sensitization of material, and high oxygen content in the water.

During the last quarter of 1974, a number of incidents of IGSCC in weld HAZs of 4-inch diameter recirculation bypass lines and in 10-inch diameter core spray lines were again observed. Following these occurrences, the NRC formed a Pipe Crack Study Group (PCSG) to (a) investigate the cause of cracks, (b) make an interim recommendation for operating plants, and (c) recommend corrective actions to be taken by future plants. The Study Group published its report (NUREG-75/067) in October 1975 which contains several recommendations to reduce the incidence of IGSCC in sensitized steel piping. Following staff review of the Study Group's recommendations, the staff issued an implementation document (NUREG-0313) which established staff positions consistent with the recommendations of the Study Group. The staff has been in the process of implementing these positions over the last couple

of years for operating plants and for plants under review for an operating license.

Since 1975, IGSCC has continued to be found in recirculation bypass and core spray lines. Incidents of IGSCC have also been observed in some stainless steel recirculation riser piping up to twelve inches in diameter and in large diameter (20 inches) recirculation piping in foreign countries. Cracks in these large recirculation lines had not been observed prior to 1975. These incidents, together with the reported questions concerning the reliability of ultrasonic inspections (UI), led to the activation of a new PCSG by NRC in September 1978.

The new Study Group was specifically chartered to reexamine the conclusions and recommendations of the 1975 PCSG report in view of cracks recently discovered in large diameter pipes. Particular attention was given to the significance of cracking found in large recirculation lines, to evaluate the capability of nondestructive examination (NDE) methods to detect IGSCC and, in addition, to assess the significance of the safe-end cracking at Duane Arnold relative to similar material and design aspects at other facilities.

The 1978 Study Group completed its evaluation and published the NUREG-0531 report in February 1979. The most important finding of this investigation was that the conclusions and recommendations reached in NUREG-75/067 by the previous PCSG and the implementation document, NUREG-0313, are still valid. The present Study Group not only reaffirmed the conclusions and recommendations reached by the previous group but also presented some new ideas to reduce the potential for IGSCC based on the operating experience since 1975 and the recent pipe cracking in large diameter pipes.

NUREG-0313, Revision 1 was published in July 1980. A complete summary of all substantive comments on the initial versions of NUREG-0313 along with their resolutions is available in the NRC public document room.

The staff is in the process of implementing its position established in the NUREG report.

Basis for Continued Plant Operation Pending Completion of Task A-42

The Staff has concluded that continued operation of Oyster Creek does not constitute an undue risk to the health and safety of the public for the following reasons:

- Although the inservice inspection programs required at Oyster Creek cannot detect all IGSCC, it is effective in locating most instances of IGSCC prior to cracks propagating through the wall.

- The leak detection system employed at Oyster Creek as a monitoring system is effective in alerting the plant operators of primary system leakage that could result from a through-wall crack.

- Sudden failure or significant loss-of-coolant is not expected from through-wall cracks prior to a period of leakage.

- Should a large through-wall crack develop, go undetected by NDE inspections, and by continuous leak detection devices, and subsequently should a rupture of the line occur causing a loss-of-coolant accident, the design of Oyster Creek is such that protection is still provided for the public health and safety.

The staff has completed its review of this task and concluded that certain surveillance requirements are appropriate (NUREG-0313, Revision 1). Compliance with these requirements will provide additional assurance that continued operation of Oyster Creek does not pose an undue risk to the health and safety of the public.

TASK A-43 CONTAINMENT EMERGENCY SUMP PERFORMANCE

Task Description

The concern addressed by this Task Action Plan for Boiling Water Reactors (BWR) is limited to the potential for degraded ECCS performance as a result of thermal insulation debris that may be blown into the suppression pool during a loss-of-coolant accident and cause backup of the pump suction line. It is planned to identify and characterize the insulation used in representative plants, estimate the quantity and nature of insulation debris resulting from pipe breaks, estimate debris distribution within the containment, and address the tolerance of safety systems and the reactor core to debris.

Basis for Continued Plant Operation Pending Completion of Task A-43

This concern is not considered significant for Oyster Creek for the following reason. Even if some insulation did reach the suppression pool, the likelihood of any insulation being drawn into an ECCS pump suction line is very small in a BWR. Suction piping to ECCS systems is typically located 4-6 feet above the pool bottom and calculated approach velocities are very low, thereby permitting debris to settle out or float on the pool surface. In addition, BWR designs employ strainers within pump suction piping and NPSH calculations for the core spray pumps are based on an assumed 50% blockage.

Accordingly, although this Task Action Plan will address insulation utilized in BWR containments, additional requirements for BWR reviews are not envisioned at this time. Therefore, we conclude that continued operation of Oyster Creek can proceed pending completion of this program without endangering the health and safety of the public.

TASK A-44 STATION BLACKOUT

Task Description

The complete loss of AC electrical power to the essential and non-essential switchgear buses in a nuclear power plant is referred to as a "Station Blackout." Because many safety systems required for reactor core decay heat removal are dependent on AC power, the consequences of a station blackout could be a severe core damage accident. Therefore, the technical issue is (a) whether the probability of a station blackout may be too high, and (b) what the consequences of a station blackout are, that is, whether severe core damage may result.

The issue of station blackout arose because of historical experience regarding the reliability of AC power supplies. A number of operating plants have experienced a total loss of offsite electrical power, and more occurrences are expected in the future. During each of these losses of offsite power events, the onsite emergency AC power supplies were available to supply the power needed by vital safety equipment. However, in some instances, one of the redundant emergency power supplies has been unavailable. In addition, there have been numerous reports of emergency diesel generators failing to start and run in operating plants.

The results of the Reactor Safety Study showed that for one of the two plants evaluated, a station blackout accident could be an important contributor to the total risk from nuclear power plant accidents. Although this total risk was found to be small, the relative importance of station blackout accidents was established. This finding and the historical diesel generator failure experience raised the concern about station blackout to an unresolved safety issue.

Basis for Continued Plant Operation Pending Completion of Task A-44

As stated in the Task Description, the purpose of this task is to evaluate the adequacy of current licensing design requirements regarding the risk of a station blackout accident resulting in unacceptable core damage.[2/] A preliminary study of operating plants was performed by the Probabilistic Analysis Staff to assess plant vulnerability using probabilistic techniques. This study did not identify any plants of unusually high susceptibility to a severe core damage accident resulting from a station blackout. To take better account of analytical uncertainties, it was decided to refine the

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[2/] In ALAB-603, the Appeal Board considered the station blackout event for St. Lucie Unit 2 and made specific findings for that facility. In addition, the Appeal Board recommended that the Commission take expeditious action to ensure that other plants and their operators are equipped to accommodate a station blackout event. This recommendation is currently under consideration by the Commission.

survey. The updated assessment is scheduled for completion in the last quarter of 1980.

In particular, the adequacy of emergency AC power supplies reliability has been questioned. The current licensing criteria require licensees to provide redundant emergency AC power supplies, to demonstrate emergency AC power supply reliability (R.G. 1.108), and include the capability of removing decay heat using at least one shutdown cooling train independent of AC power. Boiling water reactors contain various systems to remove core decay heat following the total loss of AC power. These systems at Oyster Creek consist of two isolation condensers which will provide an adequate heat sink for approximately 1.6 hours. This allows time for restoration of AC power from either offsite or onsite sources. For this reason, it is concluded that operation of the Oyster Creek facility can proceed without endangering the health and safety of the public while this task is being conducted.

PART 2

SIX ADDITIONAL ITEMS

TASK A-29 DESIGN FEATURES TO CONTROL SABOTAGE

Task Description

Extensive efforts and resources are expended in designing nuclear power plants to minimize the risk to the public health and safety from equipment or system malfunction or failure. However, reduction of the vulnerability of reactors to industrial sabotage is currently treated as a plant physical security function and not as a plant design requirement. Although present reactor designs do provide a great deal of inherent protection against industrial sabotage, extensive physical security measures are still required to provide an acceptable level of protection. An alternate approach would be to more fully consider reactor vulnerabilities to sabotage along with economy, operability, reliability, maintainability, and safety during the preliminary design phase. Since emphasis is being placed on standardizing plants, it is especially important to consider measures which could reduce the vulnerability of reactors to sabotage. Of course, any design features to enhance physical protection must be consistent with future system safety requirements.

The purpose of this task is to evaluate plant design concepts that could be implemented to achieve adequate protection against industrial sabotage without relying on extensive physical security measures. This task should result in a better understanding of design concepts that may provide alternate and more effective means of achieving adequate levels of protection. Therefore, this task does not raise a present safety concern, but seeks to augment present security systems.

Proposed research will provide the staff with (a) detailed feasibility and cost-effectiveness analysis of alternative design features for protection against sabotage, (b) impact analyses of those design features on safety, operability, reliability, and maintainability of the plant, (c) recommendations for any additional design changes which would reduce the vulnerability of reactors to industrial sabotage, and (d) evaluation of methodologies for incorporating the consideration of those design features into the review of license applications. After a review by NRC safety and safeguards staff, the recommendations of the research studies could be incorporated into revisions to the "General Design Criteria" of Appendix A of Part 50.

Basis for Continued Plant Operation Pending Completion of Task A-29

Protection of nuclear power plants against industrial sabotage is adequately accomplished through implementation of the requirements of 10 CFR 73.55. Oyster Creek has an approved security plan which has been incorporated as part of their operating license. We conclude that operation can continue prior to resolution of this generic issue without undue risk to the health and safety of the public.

TASK B-63 ISOLATION OF LOW PRESSURE SYSTEMS CONNECTED TO THE REACTOR COOLANT PRESSURE BOUNDARY

Task Description

There are several systems connected to the reactor coolant pressure boundary which have design pressures considerably below the reactor coolant system operating pressure. The Reactor Safety Study (RSS) identified a potential inter-system loss-of-coolant accident (LOCA) in a pressurized water reactor as a significant contributor to the risk of core melt. The inter-system LOCA identified in the RSS was the failure of two check valves in the injection lines of the residual heat removal system (or low-pressure injection system), that would allow the high-pressure reactor coolant to communicate with the low-pressure piping outside of containment. Rupture of the low-pressure piping could result in loss of reactor coolant outside of containment and subsequent core meltdown. On this basis, the risk-based evaluation concluded that the possible improvement in procedures for examining inter-facing system isolation devices that might result from this task, was potentially risk significant.

Basis for Continued Plant Operation Pending Completion of Task B-63

In February 1980, a letter was sent to all LWR licensees requiring an evaluation of all Reactor Coolant System Pressure isolation valves to determine if appropriate measures are being taken to ensure their integrity. Staff review (completed in August 1980) of the valve configurations extant at the Oyster Creek facility disclosed that neither of the valve configurations identified in the RSS as potential risk contributors exists at Oyster Creek. The staff considers this issue to be resolved for the Oyster Creek facility.

TASK B-30 DESIGN BASIS FLOODS AND PROBABILITY

Task Description

The purpose of this task was to detail the bases for design basis flood events used by the NRC licensing staff in case reviews and to address the possible use of probability estimates for the principal flood producing events. This task has been completed and a report to the ACRS was issued in July 1977. The report presents discussion and definitions of flood events which may be used as Design Basis Floods for review of nuclear power plants. The report supports continued use by the staff of a deterministic approach for identifying the Design Basis Flood events in preference to possible use of a probabilistic approach. The deterministic approach identifies the upper limit of flood potential physically possible. As indicated in the report, the NRC licensing staff does not feel that a probabilistic approach is appropriate for use in licensing reviews at the present time because of the lack of confidence in estimates of extreme flooding events using current techniques.

The preliminary results of the risk-based evaluation discussed indicate that the probability of a flood-induced core meltdown accident at most sites is very low. However, the study notes that detailed probabilistic estimates have not been performed for specific sites and that preliminary indications from flood data analyses performed by the NRC's Office of Nuclear Regulatory Research using probabilistic techniques indicate that flooding events could potentially be risk significant for some sites. On this basis, this issue, i.e., the use of deterministic versus probabilistic methodology for determining design basis floods, was categorized as potentially risk significant in the preliminary risk-based evaluation.

The staff has concluded that the deterministic approach is acceptably conservative and, at present, is the preferred approach for use in licensing reviews because of the large uncertainties associated with probabilistic estimates of very infrequent events.

Basis for Continued Plant Operation Pending Completion of Task B-30

A review of the flood potential at Oyster Creek will be conducted as part of the Systematic Evaluation Program (SEP). Pending completion of the SEP review, we conclude, based on the deterministic approach used by the licensee's analysis, that operation can continue without undue risk to the health and safety of the public.

TASK B-34 OCCUPATIONAL RADIATION EXPOSURE REDUCTION

Task Description

This task involves the development of additional criteria and guidelines to create improved reactor plant designs and operations to support full implementation of the NRC's regulatory requirement that radiation exposures should be maintained as low as is reasonably achievable.

Occupational radiation exposures to station and contractor personnel at operating nuclear facilities have generally been increasing over the years at both PWRs and BWRs.

General guidance with regard to maintaining occupational radiation exposures "as low as is reasonably achievable" (ALARA) is available in Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable." This guidance has been utilized by the staff in performing licensing reviews for a number of years. Task B-34 will draw from this experience and, with the aid of supplementary studies, will develop additional criteria regarding techniques and methods to maintain occupational radiation exposures ALARA.

Basis for Continued Plant Operation Pending Completion of Task B-34

Although the preliminary risk-based evaluation of this task categorized this issue as a potential high risk item, current NRC requirements and staff review procedures assure that occupational radiation exposures will be maintained as low as is reasonably achievable. Task B-34 may provide improved guidance to designers and operators for maintaining occupational radiation exposures ALARA.

We conclude that current NRC requirements and guidance on maintaining occupational radiation exposures as low as is reasonably achievable provide an acceptable basis for continued plant operation without posing an undue risk to the health and safety of the public.

TASK A-30 ADEQUACY OF SAFETY RELATED DC POWER SUPPLIES

Task Description

This generic task originated from a letter to the NRC's Advisory Committee on Reactor Safeguards from one of its consultants that questioned the reliability of DC power supplies at nuclear power stations. The specific concern expressed was as follows:

> While a nuclear power plant is operating, one of two redundant DC power supply systems fails causing a reactor scram and subsequently causing loss of all offsite power. At this point, safe shutdown of the plant requires that the residual heat from the decay of radioactivity be removed from the reactor. Control of valve position and pumps needed to remove residual heat after plant shutdown depends on availability of the DC power supply. If all remaining sources of DC power were lost, continued cooling of the reactor core cannot be assured.

Basis for Continued Plant Operation Pending Completion of Task A-30

The NRC staff's view is that the virtually simultaneous and independent failure of redundant DC power supplies is unlikely and that their failure from a common event is judged to be low enough in likelihood that adequate protection of the public health presently exists, but that additional technical studies to be provided as part of this task should and will be performed to add confidence to this judgment. This view stems from the following: (1) the postulated scenario is highly unlikely; (2) the period of vulnerability to the above cited single failure of the redundant DC power supply is limited, i.e., both the DC power supply failure initiating the scenario, and the second failure of the remaining source of DC power must occur within 30 seconds to defeat starting of the redundant diesel and acceptance of critical loads; and (3) the degree of vulnerability is mitigated substantially by the availability of alternative measures for

restoration of power or for removal of decay heat and of sufficient time (at least an hour) for operator implementation of these alternative measures.

A more detailed discussion of the design of DC power supply systems and of the NRC staff's view of the postulated accident scenario described above is provided in NUREG-0305, "Technical Report on DC Power Supplies in Nuclear Power Plants," dated July 1977. The review of DC power supplies is expected to be completed by the end of Calendar Year 1980. Any requirements resulting from this review will be considered for Oyster Creek. It is anticipated that these items will be mostly recommended improvements in maintenance and test procedures. In addition, two Systematic Evaluation Program (SEP) topics related to Oyster Creek power supplies have been completed. Although some minor deviations from current licensing criteria were noted, they were deemed not to require consideration for correction until the final SEP assessment is completed. Accordingly, the staff concludes that while this task is being performed, continued plant operation can proceed without endangering the health and safety of the public.

TASK A-15 PRIMARY COOLANT SYSTEM DECONTAMINATION AND STEAM GENERATOR CHEMICAL CLEANING

Task Description

The presence of a layer of highly radioactive corrosion products adhering to the interior surfaces of the primary coolant system has, in one case, prevented licensees from carrying out some of the less important inservice inspections required by their technical specifications. Because of the safety significance of the system and components being inspected, an approach should be developed to permit these inspections while at the same time minimizing personnel radiation exposures. Several methods of decontamination to reduce radioactivity levels in the primary system are under review by the nuclear industry for application in operating reactors. These include chemical decontamination, electropolishing, mechanical and hydraulic decontamination.

This generic task involves reviewing the existing and ongoing decontamination technology and providing guidance to the NRC staff and industry relating to acceptable methods of decontamination of reactor primary coolant systems. This task will also address the chemical cleaning of steam generators for the purpose of reducing the rate of steam generator tube degradation; however, this latter portion of this task is not applicable to boiling water reactors.

Basis for Continued Plant Operation Pending Completion of Task A-15

The radiation levels in Oyster Creek do not presently prevent the licensee from carrying out the important safety-related inspections, modifications and repairs. In addition, appropriate remote methods of inspection may be

utilized if necessary to permit inspections to be carried out. Accordingly, we conclude that there is reasonable assurance that Oyster Creek can continue to operate prior to the ultimate resolution of this generic issue without endangering the health and safety of the public.