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Report to Congress on Abnormal Occurrences

April - June 1979

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U.S. Nuclear Regulatory
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ABSTRACT

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event which the Nuclear Regulatory Commission determines to be significant from the standpoint of public health or safety and requires a quarterly report of such events to be made to Congress. This report, the seventeeth in the series, covers the period from April 1 to June 30, 1979.

The following incidents or events, including any submitted by the Agreement States, were determined by the Commission to be significant and reportable:

1. There were three abnormal occurrences at the 70 nuclear power plants licensed to operate. One involved an indication of low water level in a boiling water reactor, the second involved damage to new fuel assemblies, and the third involved deficient procedures.

2. There were no abnormal occurrences at fuel cycle facilities (other than nuclear power plants).

3. There were no abnormal occurrences at other licensee facilities.

4. There were two abnormal occurrences reported by the Agreement States. One involved releases of tritium and contamination of food and the second involved overexposures from a radiography source.

This report also contains information updating previously reported abnormal occurrences, including an update on the Nuclear Accident at Three Mile Island.

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PREFACE

INTRODUCTION

The Nuclear Regulatory Commission reports to the Congress each quarter under provisions of Section 208 of the Energy Reorganization Act of 1974 on any abnormal occurrences involving facilities and activities regulated by the NRC. An abnormal occurrence is defined in Section 208 as an unscheduled incident or event which the Commission determines is significant from the standpoint of public health or safety.

Events are currently identified as abnormal occurrences for this report by the NRC using the criteria delineated in Appendix A. These criteria were promulgated in an NRC policy statement which was published in the Federal Register (42 FR 10950) on February 24, 1977. In order to provide wide dissemination of information to the public, a Federal Register notice is issued on each abnormal occurrence with copies distributed to the NRC Public Document Room and all local public document rooms. At a minimum, each such notice contains the date and place of the occurrence and describes its nature and probable consequences.

The NRC has reviewed Licensee Event Reports, licensing and enforcement action (e.g., violations, infractions, deficiencies, civil penalties, license modifications, etc.), generic issues, significant inventory differences involving special nuclear material, and other categories of information available to the NRC. The NRC has determined that only those events, including those submitted by the Agreement States, described in this report meet the criteria for abnormal occurrence reporting. This report, the seventeenth in the series, covers the period between April 1 - June 30, 1979.

Information reported on each event includes: date and place; nature and probable consequences; cause or causes; and actions taken to prevent recurrence.

THE REGULATORY SYSTEM

The system of licensing and regulation by which NRC carries out its responsibilities is implemented through rules and regulations in Title 10 of the Code of Federal Regulations. To accomplish its objectives, NRC regularly conducts licensing proceedings, inspection and enforcement activities, evaluation of operating experience and confirmatory research, while maintaining programs for establishing standards and issuing technical reviews and studies. The NRC's role in regulating represents a complete cycle, with the NRC establishing standards and rules; issuing licenses and permits; inspecting for compliance; enforcing license requirements; and carrying on continuing evaluations, studies and research projects to improve both the regulatory process and the protection of the public health and safety. Public participation is an element of the regulatory process.

In the licensing and regulation of nuclear power plants, the NRC follows the philosophy that the health and safety of the public are best assured through the establishment of multiple levels of protection. These multiple levels can be achieved and maintained through regulations which specify requirements which will assure the safe use of nuclear materials. The regulations include design and quality assurance criteria appropriate for the various activities licensed by NRC. An inspection and enforcement program helps assure compliance with the regulations. Requirements for reporting incidents or events exist which help identify deficiencies early and aid in assuring that corrective action is taken to prevent their recurrence.

Most NRC licensee employees who work with radioactive materials are required to utilize personnel monitoring devices such as film badges or TLD (thermoluminescent dosimeter) badges. These badges are processed periodically and the exposure results normally serve as the official and legal record of the extent of personnel exposure to radiation during the period the badge was worn. If an individual's past exposure history is known and has been sufficiently low, NRC regulations permit an individual in a restricted area to receive up to three rems of whole body exposure in a calendar quarter. Higher values are permitted to the extremities or skin of the whole body. For unrestricted areas, permissible levels of radiation are considerably smaller. Permissible doses for restricted areas and unrestricted areas are stated in 10 CFR Part 20. In any case, the NRC's policy is to maintain radiation exposures to levels as low as reasonably achievable.

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REPORTABLE OCCURRENCES

Since the NRC is responsible for assuring that regulated nuclear activities are conducted safely, the nuclear industry is required to report incidents or events which involve a variance from the regulations, such as personnel overexposures, radioactive material releases above prescribed limits, and malfunctions of safety-related equipment. Thus, a reportable occurrence is any incident or event occurring at a licensed facility or related to licensed activities which NRC licensees are required to report to the NRC. The NRC evaluates each reportable occurrence to determine the safety implications involved.

Because of the broad scope of regulation and the conservative attitude toward safety, there are a large number of events reported to the NRC. The information provided in these reports is used in the NRC and the industry in their continuing evaluation and improvement of nuclear safety. Most of the reports received from licensed nuclear power facilities describe events that did not directly involve the nuclear reactor itself, but involved equipment and components which are peripheral aspects of the nuclear steam supply system, and are minor in nature with respect to impact on public health and safety. Many are discovered during routine inspection and surveillance testing and are corrected upon discovery. Typically, they concern single malfunctions of components or parts of systems, with redundant operable components or systems continuing to be available to perform the design function.

Information concerning reportable occurrences at facilities licensed or otherwise regulated by the NRC is routinely disseminated by NRC to the nuclear industry, the public, and other interested groups as these events occur. Dissemination includes deposit of incident reports in the NRC's public document rooms, special notifications to licensees and other affected or interested groups, and public announcements. In addition, a biweekly computer printout containing information on reportable events received from NRC licensees is sent to the NRC's more than 120 local public document rooms throughout the United States and to the NRC Public Document Room in Washington, D.C.

The Congress is routinely kept informed of reportable events occurring at licensed facilities.

AGREEMENT STATES

Section 274 of the Atomic Energy Act, as amended, authorizes the Commission to enter into agreements with States whereby the Commission relinquishes and the States assume regulatory authority over byproduct, source and special nuclear materials (in quantities not capable of sustaining a chain reaction). Comparable and compatible programs are the basis for agreements.

Presently, information on reportable occurrences in Agreement State licensed activities is publicly available at the State level. Certain information is also provided to the NRC under exchange of information provisions in the agreements. NRC prepares a semiannual summary of this and other information in a document entitled, "Licensing Statistics and Other Data," which is publicly available.

In early 1977 the Commission determined that abnormal occurrences happening at facilities of Agreement State licensees should be included in the quarterly report to Congress. The abnormal occurrence criteria included in Appendix A is applied uniformly to events at NRC and Agreement State licensee facilities. Procedures have been developed and implemented and any abnormal occurrences reported by the Agreement States to the NRC are included in these quarterly reports to Congress.

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REPORT TO CONGRESS ON ABNORMAL OCCURRENCES

APRIL-JUNE 1979

NUCLEAR POWER PLANTS

The NRC is reviewing events reported at the 70 nuclear power plants licensed to operate during the second quarter of 1979. As of the date of this report, the NRC had determined that the following events were abnormal occurrences.

79-5 Indication of Low Water Level in a Boiling Water Reactor

Preliminary information pertaining to this incident was reported in the Federal Register (44 FR 50925). Appendix A (Example 1 of "For Commercial Nuclear Power Plants") of this report notes that exceeding a safety limit of license Technical Specifications (10 CFR Part 50.36(c)) can be considered an abnormal occurrence.

Date and Place - On May 2, 1979, the NRC was notified by the licensee (Jersey Central Power and Light Company) of an event at their Oyster Creek facility. The Oyster Creek Nuclear Plant utilizes a boiling water reactor and is located in Ocean County, New Jersey.

Nature and Probable Consequences

Summary

A loss of feedwater transient at the Oyster Creek facility on May 2, 1979, resulted in a significant reduction in water inventory above the reactor core area as measured by one set of water level instruments (triple-low level), while the remaining two sets of level instrumentation in the reactor annulus indicated water levels above any protective feature setpoint (Figure 1). The water level measured within the core shroud area fell below the triple-low level setpoint, a safety limit, of 5-feet 6-inches above the top of the fuel. Subsequent analyses by the licensee have conservatively determined that the minimum water level over the top of the fuel was 1 to 1-1/2 feet. Coolant sample analyses and offgas release rates support the conclusion that no fuel damage occurred.

Sequence of Events

Oyster Creek is a non-jet pump BWR* with a licensed power of 1930 MWt. Immediately prior to the transient, the reactor was operating at 98% power

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*The non-jet pump BWR is of an older design. The newer designs incorporate jet pumps within the reactor pressure vessel to improve the coolant recirculation system performance. The jet pump concept reduced the number of external coolant recirculation loops to two.

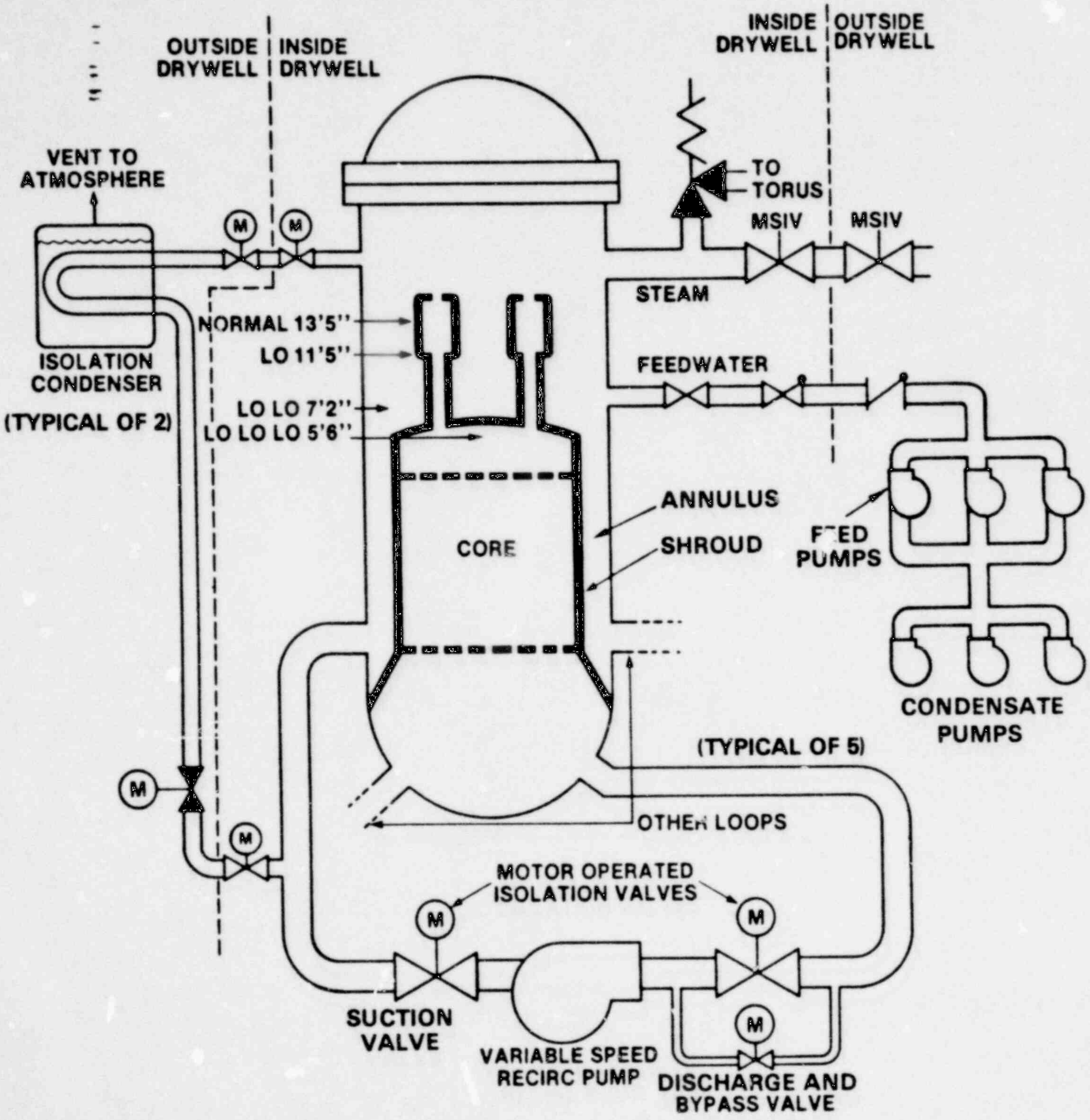

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Figure 1. RECIRCULATION, STEAM AND ISOLATION CONDENSER SCHEMATIC

with the reactor vessel water level at 13-feet 4-inches above the top of the fuel. The "D" reactor recirculation loop was out-of-service because of a recirculation pump seal cooler problem and the "SB" startup transformer was out-of-service for inspection of the associated 4160-volt cabling.

The initiating event was a false high reactor pressure scram. The pressure spike that led to the scram signal was generated by the way an instrument technician was performing surveillance testing on isolation condenser pressure switches. The signal resulted in a simultaneous reactor scram and the tripping of all operating recirculation pumps. The tripping of all operating recirculation pumps is a safeguard to mitigate the consequences of anticipated-transients-without-scram (ATWS) events.

Thirteen seconds after the reactor scram, the turbine tripped at the low load setpoint. The turbine trip initiated a transfer of power from the auxiliary transformers to the startup transformers. Because one startup transformer "SB" w s out of service, two feed pumps and two condensate pumps (pumps 1B and 1C) on the associated 4160v bus (2B) lost power. The third feed pump (1A) tripped due to low suction pressure during the feedwater transient. An immediate attempt to restart the 1A feedwater pump, powered by the live 4160v bus (1A), was unsuccessful because of failure of an auxiliary oil pump to start. The lube oil pump is interlocked in the feed pump start sequence. This was the only equipment failure during the transient.

Subsequent to the reactor scram, reactor water inventory initially decreased due to steam flow through the turbine bypass valves to the main condenser. This loss together with the void collapse associated with the scram and the subsequent loss of feed flow, resulted in a rapid reactor water level reduction to the low water level alarm setpoint of 11-feet 5-inches above the top of the fuel at 13.6 seconds. The operator manually initiated closure of all main steam line isolation valves (MSIV) at about 43 seconds into the transient to conserve water. The minimum indicated water level in the annulus was 9-feet 8-inches above the top of the fuel (the low-low setpoint is 7-feet 2-inches above the top of the fuel).

After closure of the MSIV, an isolation condenser was manually placed in service for core decay heat removal. The isolation condenser was condensing steam from the core and returning the condensate to the reactor annulus through a connection to a recirculation loop pump suction line (Figure 1). At approximately a minute and a quarter after the reactor scram, the discharge valves in "A" and "E" recirculation loops were closed in accordance with a Standing Order that was in effect. Closing the "A" and "E" loop discharge valves had in the past been necessary to prevent inadvertent stopping of the isolation condenser due to forced flow from operating recirculation pumps being sensed as if it were flow from an isolation-condenser line break. (This Standing Order was no longer appropriate since an ATWS modification had been made that tripped the recirculation pumps coincident with high-pressure or low-low-level scrams. The necessary procedure change had not been performed following the plant modification.) At the same time, the "B" and "C" loop discharge valves were apparently closed, in anticipation of restarting the recirculation pumps. The "D" loop discharge valve had been closed prior to the event because the associated pump was out of service.

The reactor triple low water level (5-feet 6-inches above the top of the fuel) setpoint was reached at 172 seconds into the transient. The triple low level setpoint activates one of the permissives on the automatic depressurization system and alarms in the control room to alert the operator.

The triple-low level in the core shroud area resulted from the restriction of the flow path between the annulus and the core region by closure of all recirculation pump discharge valves. With the recirculation pump discharge valves closed (discharge piping is over 2 feet in diameter), the only flow path back to the core region was via the 2-inch bypass lines around the discharge valves. The effect of this flow restriction was to reduce the water level in the core region and to increase the level in the reactor annulus area.

Reactor pressure was controlled by intermittent manual operation of the two isolation condensers. At about half an hour into the transient, a recirculation pump was started. The operator tripped the recirculation pump within 2 minutes when he noted a rapid decrease in the annulus level (the recirculation pumps take suction from the annulus). About 5 minutes later, a feed pump was successfully started. At about 40 minutes into the event, a primary recirculation pump and a reactor feed pump were restarted for continued cooldown of the reactor. From about 40 minutes onward, the water level within the core region was normal. The plant attained cold shutdown within 9 hours.

Review of the occurrence by the licensee and NRC established that although the water level in the core shroud area went below the triple low level setpoint, the core remained covered and consequently no fuel damage would be expected.

Cause or Causes - The spurious reactor high pressure scram was initiated by a pressure spike on the reactor high pressure scram switches, caused by an instrument technician while performing surveillance testing. However, the fundamental causes of the resulting sequence of events were:

(1) The operators essentially isolated the reactor annulus and core region from each other by shutting all recirculation pump discharge valves.

(2) Notes or Cautions against closing all suction and discharge valves in the recirculation loops were not adequate.

Actions Taken to Prevent Recurrence

Licensee - The licensee performed a thorough evaluation of the event to determine whether any fuel damage had occurred and developed follow up actions. As a result of the evaluation and discussions with the NRC, the licensee took the following significant actions:

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1. The triple-low level was established as a Safety Limit for all modes of reactor operation.[1]

2. A requirement was added to the Technical Specifications that the suction and discharge valves in at least two recirculation loops be open at all times. The procedures were changed to implement this requirement.

3. Operator training sessions were held, the event was thoroughly discussed and the revised procedures reviewed.

NRC - Following notification from the licensee of the event, an NRC inspector was dispatched to the site. Additional NRC personnel arrived at the site on May 3, 1979 to review the situation and determine the status of the plant. Fact finding by the NRC was supplemented by information obtained from the licensee, the reactor vendor (General Electric) and fuel supplier (Exxon). A safety evaluation report (SER) of the event was prepared which discusses the minimum water level experienced in the reactor vessel and the fuel conditions. The following three requirements were added to the Technical Specifications:

1. The triple-low level was made a Safety Limit for all mode-switch positions.

2. At least two recirculation loop discharge and suction valves must remain in the full open position.

3. The time duration of the low-low level signal was required to be not greater than that used in the safety analysis for the limiting loss-of-inventory transient.

The NRC staff also recommended that the licensee consider surveillance program and level instrument improvements.

It was concluded that no evidence of fuel damage was apparent, and that the facility could be safely returned to operation.

Based on the satisfactory actions taken by the licensee, on May 30, 1979 the NRC authorized the licensee to resume operation.

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[1]At the time of the event, the licensee's technical specifications defined the triple-low water level as a Safety Limit when the reactor mode switch was in the "SHUTDOWN" mode only. A limiting safety system setting was also associated with the double low water level when the mode switch was in the "RUN" position. Even though the mode switch had been placed in the "REFUEL" position by the operator shortly after initiation of the transient, the event was regarded by the licensee as if a Safety Limit had been violated.

The possible generic implications of the Oyster Creek event have been considered. Nine Mile Point Unit 1 (operated by Niagara Mohawk Power Corporation and located in Oswego County, New York) and LaCrosse (operated by Dairyland Power Cooperative and located in Monroe County, Wisconsin) are the only reactors presently operating which are susceptible to a similar event. Immediate requirements similar to those which were required for Oyster Creek (Technical Specification changes 1 and 2) were implemented at these facilities prior to their start-up (they were both in a shutdown condition at the time of the Oyster Creek event). The third requirement will be implemented as soon as practicable.

Two other plants (Dresden Unit 1 and Big Rock Point), which are presently in extended shutdowns, would also be susceptible to a similar event. However, it is planned to impose appropriate requirements on those two plants prior to their startup.

In addition, on May 29, 1979 the NRC issued IE Information Notice No. 79-13, detailing this event, to all holders of operating licenses and construction permits.

Further reports will be made as appropriate.

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79-6 Damage to New Fuel Assemblies

Preliminary information pertaining to this incident was reported in the Federal Register (44 FR 50925). Appendix A (Example 6 of "For All Licensees") of this report notes that a substantiated case of actual or attempted ... sabotage of a facility can be considered an abnormal occurrence.

Date and Place - On May 7, 1979, the NRC Resident Inspector at the Surry Power Station was notified by the licensee (Virginia Electric and Power Company - VEPCO) that while conducting inspections of new fuel for Unit 2 it was found that 62 of 64 fuel assemblies were coated with a white crystalline substance. Surry Units 1 and 2 are pressurized water nuclear power plants located in Surry County, Virginia.

Nature and Probable Consequences - On May 7, 1979, while conducting routine inspections of new fuel, the licensee discovered that a foreign substance had been poured onto 62 of the 64 new fuel assemblies stored in the Fuel Building, a vital area which contains both new and spent fuel. An analysis of the substance determined it to be sodium hydroxide. As a result of this analysis and the uncertainty of the extent of damage, the licensee is returning all the assemblies to the vendor for refurbishment. The licensee determined that there were no indications of damage to the spent fuel, nor was there evidence of unauthorized individuals gaining access to the vital area.

Fuel at the Surry site is stored in the Fuel Building, an area which is locked and alarmed, and to which access is controlled by the use of specially coded access cards. Authorized individuals, who are permitted access to the Fuel Building using the specifically coded access cards, are afforded unimpeded access to both the new and spent fuel.

Since normally conducted inspections by the licensee detected the damage to the new fuel, there is little chance that these assemblies - damaged in this way - would have been used in the reactor. While the actual consequences of this incident had no effect on the public health and safety, the incident did represent a potential threat in that it occurred within a vital area where sabotage to both new fuel and spent fuel was possible.

Cause or Causes - The cause was an alleged criminal act. On May 7, 1979, the licensee notified the FBI of the damage to the new fuel. The FBI conducted an investigation which culminated in two plant workers surrendering to Surry County authorities on June 19, 1979. A grand jury hearing was held in Surry, Virginia on July 24, 1979; trial is scheduled for October 10-12, 1979. The two workers, under advice from their attorney, have refused to describe the details of the safety issues which reportably motivated them to commit the acts.

Actions Taken to Prevent Recurrence

Licensee - As a result of the incident, and to assist the FBI in its investigation, the licensee considerably reduced the number of people permitted access to the Fuel Building and stationed a security guard inside the Fuel Building to verify access authorization. These were prompt temporary actions. The licensee has completed a thorough review of their access control program, and are now more selective in determining whether unescorted access should be provided. The licensee has made the Superintendent of Administrative Services responsible for coordinating corrective actions, and to ensure that weaknesses are corrected even if noted by someone not normally responsible for that particular professional discipline. These actions are consistent with the NRC IE Bulletin described below. Similar measures were also instituted at VEPCO's North Anna Power Station.

NRC - An NRC IE Security Inspector was dispatched to the site on May 8, 1979. Additionally, the Region II Senior Investigator, the Region II Security Section Chief and a Health Physics Inspector were onsite to assist the NRC Resident Inspector and to provide onsite assistance to the FBI. NRC IE Security Inspectors have examined the corrective measures taken by the licensee.

NRC IE Information Notice No. 79-12, "Attempted Damage to New Fuel Assemblies," was issued on May 11, 1979, to alert all NRC licensees who store new fuel assemblies of this problem.

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NRC IE Bulletin No. 79-16, "Vital Area Access Controls," was issued on July 26, 1979 to require specific actions by the licensees, including a report by September 9, 1979 of actions taken and planned.

This incident is closed for purposes of this report.

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79-7 Deficient Procedures

(During preparation of this report, the following item was determined reportable, using the criteria given in Appendix A of this report. Example 11 (For All Licensees) notes that serious deficiency in management or procedural controls in major areas can be considered an abnormal occurrence. Federal Register noticing is being made in conjunction with the noticing of issuance of this report.)

Date and Place - On June 2, 1979 an NRC inspector discovered a condition which indicated a deficiency in procedural controls at Arkansas Nuclear One-Unit 1. The unit utilizes a pressurized water reactor and is located in Pope County, Arkansas.

Nature and Probable Consequences - On June 2, 1979 while Arkansas Nuclear One-Unit 1 was preparing for startup, an NRC inspector in the control room found the controls of the emergency feedwater system positioned so that the system could not automatically respond if needed. There was no assurance that the system would have been returned to its normal standby status prior to power operation had not the inspector noticed the problem, since there was no procedural requirement to check the system status.

The emergency feedwater controls had been placed in the improper position when licensed operators were performing a surveillance test of the main feedwater check valves. The surveillance procedure did not include, as it should have, instructions to bypass the emergency feedwater system and return it to normal standby status at the end of the test. The operators bypassed the emergency feedwater system on their own initiative and had not returned it to normal at the time the NRC inspector noticed the condition.

The improper positioning of the emergency feedwater controls did not represent an immediate safety hazard. Operators had recently been required by the NRC to receive training in manual initiation of the emergency feedwater system as a result of the experience of the Three Mile Island accident. However, the NRC staff was concerned that procedural inadequacies and operator-initiated deviation from written procedures allowed such a situation to develop and that more general problems with procedural controls might exist at the Arkansas plant.

Cause or Causes - Licensed operators deviated from written procedures, placing a safety-related system in a condition which would be unsafe during power operation and there were no procedural checks which would have assured that this condition would be corrected prior to power operation.

Actions Taken to Prevent Recurrence

Licensee - Arkansas Power and Light Company returned the plant to cold shutdown. In compliance with a June 2, 1979 NRC confirmatory order, the licensee maintained the plant in cold shutdown until the NRC staff was satisfied with utility methods for controlling the development of operating procedures, the adequacy of existing procedures and until there was assurance that operators would not deviate from those procedures.

NRC - The NRC issued an Order on June 2, 1979 confirming the requirement for cold shutdown while procedural controls and operator adherence to written procedures were re-examined. NRC inspectors confirmed the adequacy of the licensee's response to the requirements of the Order. On June 14, 1979, Arkansas Power and Light Company was authorized to return Arkansas Nuclear One-Unit 1 to operation.

IE Information Notice No. 79-15 ("Deficient Procedures") was issued on June 7, 1979 to all holders of reactor operating licenses and construction permits to inform them of this event.

This incident is closed for purposes of this report.

FUEL CYCLE FACILITIES

(Other Than Nuclear Power Plants)

The NRC is reviewing events reported by these licensees during the second quarter of 1979. As of the date of this report, the NRC had not determined that any events were abnormal occurrences.

OTHER NRC LICENSEES

(Industrial Radiographers, Medical Institutions, Industrial Users, etc.)

There are currently more than 8,000 NRC nuclear material licenses in effect in the United States, principally for use of radioisotopes in the medical, industrial and academic fields. Incidents were reported in this category from licensees such as radiographers, medical institutions, and byproduct material users.

The NRC is reviewing events reported by these licensees during the second quarter of 1979. As of the date of this report, the NRC had not determined that any events were abnormal occurrences.

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AGREEMENT STATE LICENSEES

Procedures have been developed for the Agreement States to screen unscheduled incidents or events using the same criteria as the NRC (see Appendix A) and report the events to the NRC for inclusion in this report. During the second quarter of 1979, the Agreement States reported the following abnormal occurrences to the NRC.

AS79-1 Releases of Tritium and Contamination of Food

Appendix A (Example 11 "For All Licensees") of this report notes that a serious deficiency in management or procedural controls in major areas can be considered an abnormal occurrence.

Date and Place - On March 9, 1979, the Arizona Atomic Energy Commission conducted an inspection at American Atomics Corporation in Tuscon, Arizona. American Atomics is licensed by Arizona to, among other things, manufacture and distribute to authorized persons luminous signs and devices using tritium, a radioactive isotope of hydrogen, as the activating agent. The inspection disclosed 4 items of non-compliance which were reported to the licensee by letter dated March 30, 1979: discharge of tritium to the atmosphere in unrestricted areas in concentrations which exceed regulatory limits; possession of tritium by the licensee in quantities in excess of that authorized by the license; inadequate stack monitoring; and the excessive use of the category of normal operating losses for accountability of tritium. The inspection report noted that in the second quarter of 1978, 57,417 curies were calculated as "normal operating loss" and as much as 80% of this was discharged to the atmosphere. For the calendar year, the 287,000 curies categorized as "normal operating losses" were deemed by Arizona to be excessive.

An unannounced investigation by the State performed on May 7, 1979 disclosed the licensee had received additional quantities of tritium and its inventory continued to exceed the amount authorized by its license.

In May 1979, the State collected environmental samples around the facility. Analyses of these samples disclosed elevated levels of tritium. Located a block from the licensee is a kitchen that prepares school lunches for the Tuscon Unified School District. Food samples were collected and the results of the analyses were reported to the State on May 31, 1979. Elevated levels of tritium were found - water contained in cake contained 56 nanocuries of tritium per liter. As a comparison, the EPA drinking water standard is 20 nanocuries per liter. At present there is no standard for tritium contained in foodstuffs. The Acting Executive Director of the Arizona Atomic Energy Commission chose to apply the EPA drinking water standard for tritium. An EPA radiation protection expert confirmed at the time that the use of that value was an acceptable standard.

Nature and Probable Consequences - American Atomics Corporation was authorized by the Arizona Atomic Energy Commission to use radioactive materials in research and development and in manufacturing of devices containing radioactive materials. Predominant among its licensed activities is the manufacture of sealed self-luminous devices containing tritium. Examples are small tubes containing up to 200 millicuries of tritium used for backlighting liquid crystal display digital watches and Exit Signs containing up to 21 curies of tritium. The former can be distributed to the public as assembled timepieces exempt from licensing only under authority of a specific license issued by NRC. The latter can be distributed to persons who possess them under a General License provided by 10 CFR Part 31.5 or equivalent Agreement State regulations.

After receiving the notice of violations from the State dated March 30, 1979, the licensee established the boundary of the restricted area at the plant boundary and the State calculated that the concentrations of tritium at this boundary did not exceed the limits prescribed for unrestricted areas. The State, however, became concerned over the consequences of the releases to the atmosphere of the operational losses.

After consultation with EPA, an environmental sampling and analysis program commenced which led to the finding on May 31, 1979, of tritium in foodstuffs used in the Tucson school system.

Assessments of doses received by the public and the health effects resulting from the releases of the tritium and the contamination of the foodstuffs will be made by the University of Arizona Health Sciences Center. It should be noted that a daily intake by an adult of two liters of water containing tritium at a concentration equal to the EPA drinking water standard will result in an annual whole body dose of 4 millirem.

Cause or Causes - It appears that the primary cause of the contamination of foodstuffs was failure by the licensee to institute managerial and procedural controls to keep releases of tritium to the atmosphere as low as reasonably achievable.

Actions Taken to Prevent Recurrence

Licensee - The licensee suspended operations on June 15, 1979 and will decontaminate and decommission the facility in Tucson.

Arizona Atomic Energy Commission (AAEC) - In collaboration with the Pima County Health Department, AAEC obtained agreement from the school district kitchen on June 1, 1979, to suspend operation until additional measurements were made.

On June 2, 1979, AAEC met and determined an emergency existed and moved to restrict the licensee's operations from two to one shift per day, and scheduled a formal hearing for June 16, 1979, to consider alteration, suspension or revocation of the license. On June 15, 1979, AAEC ordered shutdown of the tritium operations. On July 11, 1979, American Atomics Corporation was given 100 days to decommission their operations. All production of tritium products was terminated and all tritium and tritium containing products were sealed to assure compliance with the Order.

On September 11, 1979, the production remnants, consisting of both rejected and leaking small tubes, as well as unfinished production items, totaling approximately 4 million pieces, were transferred to 17H-55 gallon drums and the drums were sealed gas tight. This effectively reduced the releases to the atmosphere to only out gassing from the production machinery, structural components, etc. This reduced the total release to approximately 4 curies per day. When the drums are connected in the total containment system, the total release is estimated to be as low as 1 millicurie per day.

On September 26, 1979 Arizona Governor Bruce Babbitt ordered the National Guard to package and transport the tritium to the Navajo Ordinance Depot. The State has signed a lease with the Department of the Army for storage of the tritium for a 60-day period from September 28, 1979 to November 26, 1979.

Future reports will be made as appropriate.

AS79-2 Overexposures from a Radiography Source

Dates and Place - On June 22, 1979, the Radiological Health Section of the State of California was notified that a possible exposure to persons occurred from a radiography incident. The State investigation revealed the following: On May 22, 1979, X-Ray Products Corporation conducted radiography at the plant of REPCO, a pressure vessel manufacturer. The radiographer made 2 or 3 exposures and, unknown to the radiographer, the source had disconnected and was found on the floor by a REPCO employee who placed it in his hip pocket. The industrial radiographer had not performed a radiation survey. Several hours later he gave it to his supervisor. Both handled it and it was left with a secretary who was asked to contact the radiographer. The radiographer returned and retrieved the source. However, the radiographer did not inform the individuals (a total of nine individuals were exposed unnecessarily), nor report the event to responsible management within his own and the customer's company. On the evening of May 22, the REPCO employee who had picked up the source became nauseous and went to a hospital where a blister was found on his buttock. The initial diagnosis and treatment was for an insect bite.

Nature and Probable Consequences - On June 22, 1979, the individual was hospitalized for treatment of injury. At that time he asked the physician if there was any relationship of the injury to the radiography performed at the plant on May 22, 1979. This worker experienced blistering on his buttocks and required surgical repair of the ulcerated skin in the region of the buttock and hip. The individual has since been released from the hospital and his progress is being monitored by the attending physician. At the time the State was notified, exposure estimates ranged from: 1st REPCO employee 1.5 million Rem surface dose, 1 cm depth dose 60,000 rem, 3 cm depth dose 7,000 rem; Supervisor 3000 to 5000 rem hand dose and 16 rem whole body; secretary 1000 to 2000 rem hand dose and 50 to 60 rem whole body. Several other workers and clerical staff were identified as receiving exposure and their dose estimates range from a high of 14 rem to 3 rem whole body.

In addition to the individual who placed the source in his hip pocket (individual A), two other individuals who handled the sources displayed clear evidence of radiation burns on their fingers. In all cases, apparently normal skin has returned. In the future, these tissues may be unusually sensitive to trauma or stress and there is a risk of late sequelae such as deterioration of the tissues.

With the exception of individual A, major systemic effects indicative of a significant whole body dose (greater than 50-100 rem) were not observed for any of the other individuals. The whole body doses incurred carry a very small increase in the statistical risk of late effects, from exposure to ionizing radiation, including cancer. All information concerning this incident was obtained from California officials.

<u>Cause or Causes</u> - This accident is under review by a State Board of Inquiry and it is premature at this time to determine possible causes until the State inquiry is completed. When determined, the causes will be identified in a future update to this report.

<u>Actions Taken to Prevent Recurrence</u> - The State issued an order on June 28, 1979 to X-Ray Products suspending their license and the investigation is continuing. The State is convening a State Board of Inquiry for this incident. The NRC issued IE Circular No. 79-16 notifying radiography licensees of the seriousness of this case and alerting them to improve training and retraining with special emphasis on performing radiation surveys and prompt notification to management of unusual events. Copies were also sent to the Agreement States for distribution to Agreement State industrial radiography licensees.

Future reports will be made as appropriate.

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APPENDIX A

ABNORMAL OCCURRENCE CRITERIA

The following criteria for this report's abnormal occurrence determinations were set forth in an NRC policy statement published in the Federal Register (42 FR 10950) on February 24, 1977.

Events involving a major reduction in the degree of protection of the public health or safety. Such an event would involve a moderate or more severe impact on the public health or safety and could include but need not be limited to:

1. Moderate exposure to, or release of, radioactive material licensed by or otherwise regulated by the Commission;

2. Major degradation of essential safety-related equipment; or

3. Major deficiencies in design, construction, use of, or management controls for licensed facilities or material.

Examples of the types of events that are evaluated in detail using these criteria are:

For All Licensees

1. Exposure of the whole body of any individual to 25 rems or more of radiation; exposure of the skin of the whole body of any individual to 150 rems or more of radiation; or exposure of the feet, ankles, hands or forearms of any individual to 375 rems or more of radiation (10 CFR Part 20.403(a)(1)), or equivalent exposures from internal sources.

2. An exposure to an individual in an unrestricted area such that the whole body dose received exceeds 0.5 rem in one calendar year (10 CFR Part 20.105(a)).

3. The release of radioactive material to an unrestricted area in concentrations which, if averaged over a period of 24 hours, exceed 500 times the regulatory limit of Appendix B, Table II, 10 CFR Part 20 (10 CFR Part 20.403(b)).

4. Radiation or contamination levels in excess of design values on packages, or loss of confinement of radioactive material such as (a) a radiation dose rate of 1,000 mrem per hour three feet from the

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surface of a package containing the radioactive material, or (b) release of radioactive material from a package in amounts greater than the regulatory limit (10 CFR Part 71.36(a)).

5. Any loss of licensed material in such quantities and under such circumstances that substantial hazard may result to persons in unrestricted areas.

6. A substantiated case of actual or attempted theft or diversion of licensed material or sabotage of a facility.

7. Any substantiated loss of special nuclear material or any substantiated inventory discrepancy which is judged to be significant relative to normally expected performance and which is judged to be caused by theft or diversion or by substantial breakdown of the accountability system.

8. Any substantial breakdown of physical security or material control (i.e., access control, containment, or accountability systems) that significantly weakened the protection against theft, diversion or sabotage.

9. An accidental criticality (10 CFR Part 70.52(a)).

10. A major deficiency in design, construction or operation having safety implications requiring immediate remedial action.

11. Serious deficiency in management or procedural controls in major areas.

12. Series of events (where individual events are not of major importance), recurring incidents, and incidents with implications for similar facilities (generic incidents), which create major safety concern.

For Commercial Nuclear Power Plants

1. Exceeding a safety limit of license Technical Specifications (10 CFR Part 50.36(c)).

2. Major degradation of fuel integrity, primary coolant pressure boundary, or primary containment boundary.

3. Loss of plant capability to perform essential safety functions such that a potential release of radioactivity in excess of 10 CFR Part 100 guidelines could result from a postulated transient or accident (e.g., loss of emergency core cooling system, loss of control rod system).

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4. Discovery of a major condition not specifically considered in the Safety Analysis Report (SAR) or Technical Specifications that require immediate remedial action.

5. Personnel error or procedural deficiencies which result in loss of plant capability to perform essential safety functions such that a potential release of radioactivity in excess of 10 CFR Part 100 guidelines could result from a postulated transient or accident (e.g., loss of emergency core cooling system, loss of control rod systems).

For Fuel Cycle Licensees

1. A safety limit of license Technical Specifications is exceeded and a plant shutdown is required (10 CFR Part 50.36(c)).

2. A major condition not specifically considered in the Safety Analysis Report or Technical Specifications that requires immediate remedial action.

3. An event which seriously compromised the ability of a confinement system to perform its designated function.

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APPENDIX B

UPDATE OF PREVIOUSLY REPORTED ABNORMAL OCCURRENCES

During the April through June 1979 period, the NRC, NRC licensees, Agreement States, Agreement State licensees, and other involved parties, such as reactor vendors and architects and engineers, continued with the implementation of actions necessary to prevent recurrence of previously reported abnormal occurrences. The referenced Congressional abnormal occurrence reports below provide the initial and any updating information on the abnormal occurrences discussed. Those occurrences not now considered closed will be discussed in subsequent reports in the series.

NUCLEAR POWER PLANTS

The following abnormal occurrence was originally reported in NUREG-75/090, "Report to Congress on Abnormal Occurrences: January-June 1975," and updated in subsequent reports in this series, i.e., NUREG-0090-1, 2, 3, 9, and Vol. 1, No. 3. It is further updated as follows:

75-5 Cracks in Pipes at Boiling Water Reactors (BWRs)

The 1978 Study Group completed its evaluation in February 1979 and issued a report, NUREG-0531, "Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants." The new Study Group not only reaffirmed the conclusions and recommendations reached by the previous group (NUREG-75/067) but also presented some new ideas to reduce the potential for intergranular stress corrosion cracking (IGSCC). In addition, they addressed IGSCC in safe ends.

On March 13, 1979, NRC issued a Notice in the Federal Register to request public comment on the Study Group's report, NUREG-0531. After expiration of the public comment period and review of the Study Group's conclusions/recommendations, the staff initiated action in June 1979 to update NUREG-0313 ("Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," issued July 1977) to incorporate the present Study Group's conclusions/recommendations and public comments received on NUREG-0531.

The NRC staff is currently updating the implementation document NUREG-0313 as a subtask under Generic Task A-42, "Pipe Cracks in Boiling Water Reactors." The objective of other subtasks is to identify and recommend additional measures to reduce the susceptibility of stainless steel piping to stress corrosion cracking. A report on the results of this task is expected to be published this year.

Further reports will be made as appropriate.

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The following abnormal occurrence was originally reported in NUREG-75/090, "Report to Congress on Abnormal Occurrences: January-June 1975," and updated in subsequent reports in this series, i.e., NUREG-0090-1, 6, and Vol. 1, No. 4. It is further updated as follows:

75-7 <u>Steam Generator Feedwater Flow Instability at Pressurized Water Reactors (PWRs)</u>

Since the previous 1978 update of this item (NUREG-0090, Vol. 1, No. 4), additional incidents of steam generator water hammer have occurred at two pressurized water reactors. These events occurred at Zion Unit 1 and San Onofre Unit 1. One event at Zion Unit 1 resulted in the actuation of the Safety Injection System. Subsequent visual inspection of the piping at Zion indicated no apparent structural damage as a result of the water hammer and safety injection. The feedwater pipes were also radiographically inspected in the vicinity of the steam generator nozzle and the pipes were found to have no cracks. Operation of the Zion unit was resumed but the licensee will modify the feedrings in all steam generators of Units 1 and 2 at the rate of one steam generator per refueling outage in order to prevent steam generator water hammer in the future. The water hammer at San Onofre Unit 1 resulted in minor damage to a seismic snubber, which was repaired.

Steam generator water hammer has occurred in certain nuclear power plants as a result of the rapid condensation of steam in a steam generator feedwater line. The consequent acceleration of a slug of water and the impact ("hammering") within the piping system causes undue stresses in the piping and its support system. The significance of these events varies from plant to plant. Since the total loss of feedwater could affect the ability of the plant to cool down after a reactor shutdown, the NRC is concerned about these events occurring, even though an event with potentially serious consequences is unlikely to happen.

With the exception of the Zion and San Onofre Units, all operating nuclear power stations that have had steam generator water hammer events have been modified and subsequently have not experienced steam generator water hammer. The NRC is continuing to evaluate the potential for steam generator water hammer in operating pressurized water reactor systems.

Further reports will be made as appropriate.

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76-16 <u>Feedwater Nozzle Cracking in Boiling Water Reactors</u>

Over the last several years, inspections at 22 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 18 of those facilities inspected. One facility has not yet been inspected because it has not accumulated significant operating time. In a closely related area, cracks have been found in control rod drive (CRD) return line nozzles, 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. The cracks resemble those found in feedwater nozzles. Both conditions probably result from the same kind of cyclic thermal stresses.

The NRC staff has completed its review of the proposed long-term solutions to the BWR nozzle cracking problem and has concluded that they provide effective means of mitigating the problem. These solutions include removal of the stainless steel cladding, replacement of existing feedwater spargers with a triple-sleeve sparger design, and a number of possible changes to feedwater systems and methods of operating the reactor. A NUREG document is being written to incorporate guidance for operating reactors and plants under licensing review. The resolution of inservice inspection technique selection and frequency of inspection has been separated from the generic task while major industry investigations (including thermal cracking of a full-size nozzle mockup for use in ultrasonic testing evaluation) continue. A revision to the NUREG document will be written at the completion of these studies. In the meantime, stringent inspection requirements, based mainly upon dye-penetrant testing, are still in force. All licensee efforts, such as system and operational changes, to lengthen the time to crack initiation and to slow crack growth are taken into account in the determination of inspection techniques and acceptance criteria. Plant modifications related to final resolution of the CRD nozzle problem are still under NRC staff review.

Plant-specific implementation of the generic resolution (with the exception of final inservice inspection technique and frequency determination) has begun.

Further reports will be made as appropriate.

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The following abnormal occurrence was originally reported in NUREG-0090-10, "Report to Congress on Abnormal Occurrences: October-December 1977," and updated in subsequent reports in this series, i.e., NUREG-0090, Vol. 1, No. 1 and Vol. 1, No. 2. It is further updated as follows:

77-9 <u>Environmental Qualification of Safety-Related Electrical Equipment Inside Containment</u>

As described in the last update to this abnormal occurrence (NUREG-0090, Vol. 1, No. 2), Inspection and Enforcement (IE) Circular No. 78-08 was issued on May 31, 1978 to all licensees to highlight important lessons learned from environmental qualification deficiencies reported by individual licensees. Licensees were requested to examine installed safety-related electrical equipment and determine that proper documentation existed which provided assurance that the equipment would function under postulated accident conditions.

NRC inspections conducted of licensees' activities in response to the Circular identified one component (certain stem mounted limit switches) found to be unqualified for service within the Loss of Coolant (LOCA) environment. Also, NRC inspection of component qualification identified equipment which did not have documentation indicating it was qualified for the LOCA environment. The inspections also identified that the licensees' re-review and resolution of problem areas were not receiving the level of attention from all licensees that the NRC believed was warranted. Therefore, IE Bulletin No. 79-01 was issued to licensees of power reactor facilities on February 8, 1979. The intent of the Bulletin was to raise the threshold of the Circular to the level of a Bulletin; i.e., actions requiring licensee response.

In addition to requiring a complete review by the licensees of the enviornmental qualification of all Class IE electrical equipment within 120 days, the Bulletin also required that any equipment determined to be unqualified for its service conditions be reported to the NRC Director of the Division of Operating Reactors within 24 hours of discovery. To date, there have been some 32 separate reports of unqualified equipment at 29 different plants involving five different types of equipment. The unqualified equipment reported included: (1) limit switches mounted on safety-related valve stems to indicate valve stem position; (2) containment isolation valve motor operators; (3) instrument and control cable insulated terminal lugs; (4) aluminum limit switch housings on containment isolation valves; and (5) ASCO pilot solenoid valves for miscellaneous valve air operators.

In each instance where an item of equipment was determined to be unqualified, the NRC staff immediately evaluated the impact on the health and safety of the public and the adequacy of the remedial steps to be taken by the licensees. In some cases the licensees elected to replace the unqualified equipment immediately; in others a basis for continued operation pending corrective action at a specified future date was provided. In those cases where the licensees proposed to continue to operate the plant for a period of time before shutting down and replacing the affected equipment, the following factors were considered in the NRC staff evaluations of whether the plants could continue to be operated safely: (1) redundant/diverse components available to perform the required safety functions; (2) locking the affected component in its safety position; (3) administrative actions and revised operating procedures; (4) additional operability tests and inspections; (5) post accident mitigating actions available; and (6) fail safe design features. In all cases where continued operation was requested by the licensees based on a plant specific safety evaluation, the NRC staff has concluded (contingent upon additional staff requirements being satisfied in some cases) that the plants could continue to be operated safely.

An NRC task group has been formed to review in detail the licensees' responses to Bulletin 79-01. The reviews will be conducted in accordance with guidelines being prepared specifically for evaluating the qualifications of Class IE equipment in operating reactors. The guidelines will address all of the significant aspects of the most current industry standard for Class IE electrical qualification, IEEE Std. 323-1974.

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Further reports will be made as appropriate.

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The following abnormal occurrence was originally reported in NUREG-0090, Vol.1, No. 2, "Report to Congress on Abnormal Occurrences: April-June 1978," and updated in NUREG-0090, Vol. 1, No. 4. It is further updated as follows:

78-2 Fuel Assembly Control Rod Guide Tube Integrity (A Generic Concern)

As reported previously, examination of fuel assembly control rod guide tubes after service in several operating pressurized water reactors (PWRs) disclosed significant amounts of wear. At the extreme, some tubes had been worn through showing sizeable holes. The cause was determined to be flow-induced vibration of fully withdrawn control rods. The rod tips, vibrating against the guide tubes, induced degrading wear, probably aided by corrosion.

The safety significance of the incidents relates to the functions of the guide tubes. Guide tubes serve both as fuel assembly structural members and as channels for control rod movement. Thus, guide tube failure could adversely affect either the preservation of a coolable core geometry or the scram capability of the control rods, or both.

The observed severe wear of the guide tubes thus far has been confined to facilities designed by Combustion Engineering (CE). Basic differences in the design of the control rod systems which insert into the guide tubes of the fuel assemblies exists between the CE plants and the other PWR plants (Westinghouse and Babcock and Wilcox). These design differences appear to have reduced the severity of wear on the guide tubes in the latter vendors facilities. However, such wear in Westinghouse and Babcock and Wilcox plants and in Exxon Nuclear fuel assemblies is under investigation by the NRC staff.

To overcome the susceptibility to wear by the guide tube material (Zircaloy-4) and to recover the design margin lost by wear, CE designed stainless steel sleeves for use in the guide tubes. Prior to installation of stainless steel sleeves during a refueling outage, operators of CE reactors instituted the practice of inserting the control rods three inches further into the core than the normal fully withdrawn position. That action both distributed the wear location and provided added assurance of scram capability. NRC approval was granted for this short-term administrative procedure allowing continued operation with the control rods inserted three inches further into the core.

The use of sleeved guide tubes was approved by the NRC as an interim repair to mitigate the guide tube wear on a cycle specific basis. In conjunction with the use of the stainless steel sleeves, the NRC staff required that inspection programs be submitted for review and approval well in advance of refueling shutdowns.

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The first opportunity to evaluate the performance of the sleeved guide tubes after reactor operations occurred during the Millstone Unit 2 refueling outage in the spring of 1979. Subsequent to the Millstone 2 refueling, the St. Lucie Unit No. 1 and the Calvert Cliffs Unit No. 1 also provided additional evidence on the performance of the sleeved guide tubes. Based on the results of these inspections, the sleeving modification has performed well as an interim solution to mitigate the guide tube wear, but it does not eliminate the cause of the wear.

Additional out-of-reactor hot loop testing by CE showed the important role of flow-induced vibration of the control rods in the guide tube wear problem. The vibration and, hence, the wear, was reduced by redistributing some of the guide tube coolant (water) flow. Two fuel assembly modifications were designed to redistribute the coolant flow. One involved inserting a splined cylinder in the top of the guide tube. The second involved reducing the size and number of flow holes in the bottom of the guide tube. Test results favored the modified flow hole design. A limited number of assemblies with both modifications are installed in currently operating reactors to confirm the loop test results.

The Calvert Cliffs Unit No. 2 is scheduled for refueling in the late summer or early fall of 1979. This unit has, in addition to the sleeving modification, 16 assemblies with modifications designed to affect the coolant flow and perturb the vibrational characteristics of the control rods.

The NRC has closely monitored the analyses and experiments performed by CE. The NRC staff agrees with the vendor that the results point to control rod flow-induced vibration as the principal factor in guide tube wear. Therefore, design modifications intended to redistribute flow in guide tubes were judged appropriate. The NRC has approved the modified designs for limited operation on the basis that they will mitigate the wear problem. Approval of either design modification as a final solution to the problem will be contingent upon the results of further out-of-reactor experiments and examination of the modified assemblies which are currently subject to in-reactor operations.

Further reports will be made as appropriate.

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The following abnormal occurrence was originally reported in NUREG-0090, Vol. 1, No. 4, "Report to Congress on Abnormal Occurrences: October-December 1978," and is further updated as follows:

78-5 <u>Loss of Containment Integrity</u>

The NRC staff review of the generic implications of the two events (Millstone Unit 2, and Salem Unit 1), has continued. The NRC staff letter of November 1978 has requested all licensees of operating reactors to respond to generic

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concerns about containment purging or venting during normal plant operation. The generic concerns were twofold:

1. Events had occurred where licensees overrode or bypassed the safety actuation isolation signals to the containment valves.

2. Recent licensing reviews have required tests or analyses to show that containment purge or vent valves would shut without degrading containment integrity during the dynamic loads of a design basis accident-loss of coolant accident (DBA-LOCA).

The NRC position of the November 1978 letter requested that licensees take the following positive actions pending completion of the NRC review: (1) prohibit the override or bypass of any safety actuation signal which would affect another safety actuation signal; the NRC Office of Inspection and Enforcement would verify that administrative controls prevent improper manual defeat of safety actuation signals, and (2) cease purging (or venting) of containment or to limit purging (or venting) to an absolute minimum, not to exceed 90 hours per year. Licensees were requested to demonstrate (by test and analysis) that containment isolation valves would shut under postulated DBA-LOCA conditions.

After the licensee responses were received for review, the NRC staff made site visits to several facilities, met with other licensees at Bethesda, Maryland, and held numerous conferences with many other licensees. The staff also met with some valve manufacturers. During these discussions the staff stressed that positive actions must be taken to assure that containment integrity would be maintained in the event of a DBA-LOCA.

As a result of these actions, the NRC staff was informed that at least three valve vendors have reported that their valves may not close against the ascending differential pressure and the resulting dynamic loading of a design basis LOCA. All identified licensees whose plants had questioned the designs are maintaining the valves in the closed position or are restricting the opening of the valves when primary containment integrity is required. Re-evaluation of the valve performances under the DBA-LOCA condition are being made by affected licensees.

At this time, the licensees of about twenty percent of the reactors have not yet limited purging and venting of containment beyond their current licensed requirements. The remainder of the licensees have either ceased purging (about twenty-five percent of the reactors) or have limited purging to various degrees. As the NRC review progresses, licensees which might have electrical override circuitry problems are being advised not to use the override and have taken compensatory interim measures to minimize the problem. The NRC is continuing to take such action during the remaining reviews.

Pending completion of the NRC staff's review, the following interim measures will be required by licensees of operating reactors that do not now limit purging or venting of containment. These licensees will be required to

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propose modifications to plant or system design to minimize the need for purging or venting of the containment. Design modifications being considered include limiting valve angular opening to assure that critical valve parts will not be damaged during the DBA-LOCA, increasing the cooling capacity of the containment cooling system to control the containment pressure, temperature and relative humidity, and using internal charcoal filter system for air circulation and filtering throughout all containment and during plant discharge to reduce airborne activity.

Further reports will be made as appropriate.

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The following abnormal occurrence was originally reported in NUREG-0090, Vol. 2, No. 1, "Report to Congress on Abnormal Occurrences: January-March 1979," and is further updated as follows:

79-1 <u>Degraded Engineered Safety Features</u>

As described in NUREG-0090, Vol. 2, No. 1, three safety concerns emerged from the analysis of the event that occurred at Arkansas Nuclear One (ANO) site on September 16, 1978. The three concerns were:

1. The offsite power supply for ANO Unit 1 Engineered Safety Feature loads was deficient in that degraded voltage could have resulted in the unavailability of ESF equipment, if it were to be needed.

2. The design of the ANO site electrical system that provides offsite power to Units 1 and 2 did not fully meet the Commission's Regulations, 10 CFR 50, Appendix A, General Design Criterion 17, because in certain circumstances a failure of one of the two offsite power circuits would also result in a failure of the other such circuit.

3. Deficiencies existed in the operation of the Unit 2 inverters that convert battery power to AC power for certain safety-related equipment.

As stated in the previous report, the NRC has reviewed and approved corrective actions taken by the licensee to prevent recurrence. The actions taken to date have satisfactorily alleviated the safety concerns (2) and (3) above. The review and evaluation by NRC staff of corrective actions proposed by the licensee addressing safety concern (1) is still in progress.

The existing NRC generic review activity regarding degraded grid voltage related to the July 5, 1976 Millstone Unit 2 event* has been expanded to ensure that adequate voltage will be available at the ESF buses during all electrical transients including voltage degradation resulting from loading

*Reference Abnormal Occurrence No. 76-9 ("Failure of Undervoltage Trip Logic and Consequent Loss of Safeguard Power") reported in NUREG-0090-5 and NUREG-0090-6.

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due to onsite automatic switching. A letter was sent from the NRC to power reactor licensees on August 8, 1979 requesting the licensees to review the adequacy of their electric power systems. Responses were requested within 60 days.

Further reports will be made as appropriate.

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The following abnormal occurrence was originally reported in NUREG-0090, Vol. 2, No. 1, "Report to Congress on Abnormal Occurrences: January-March 1979," and is further updated as follows:

79-2 <u>Deficiencies in Piping Design</u>

The Nuclear Regulatory Commission staff ordered five plants to shutdown on March 13, 1979, until reanalysis and necessary modifications were made to safety-related piping systems to bring them into conformance with requirements for withstanding earthquakes. The plants ordered shutdown were Beaver Valley Unit 1, James A. FitzPatrick, Maine Yankee and Surry Units 1 and 2.

Stone and Webster Engineering, the architect engineer for all five plants, and Duquesne Light Company, the licensee for the Beaver Valley facility, reported to the NRC during a meeting on March 8, 1979, that an algebraic summation method was used to combine seismic forces in the computer code SHOCK II. The algebraic summation method can result in cancellation of seismic forces and resulted in prediction of stresses significantly lower than would be predicted by NRC approved techniques. Following the meeting on March 8, members of the NRC staff met for three days with Stone & Webster Engineering officials in Boston. Additional analyses of piping systems for the Beaver Valley facility were performed. These analyses indicated significant overstress in the piping systems under postulated earthquake conditions when computer codes were utilized which did not combine seismic loads algebraically. Piping systems involving the integrity of the reactor coolant pressure boundary, Emergency Core Cooling Systems and safe shutdown systems, were involved. It was also determined that the same computer code (SHOCK II) was used in the design of four other facilities. The NRC staff ordered all five plants shut down because there was not assurance that a severe earthquake at any of these facilities would not cause an accident, damage emergency core cooling systems, and prevent safe shutdown of the plant.

The required reanalysis and necessary modifications were completed for Maine Yankee and Beaver Valley and orders were issued on May 24, 1979 and August 8, 1979, respectively terminating the March 13, 1979 Show Cause Orders. Sufficient reanalysis and modifications were completed for FitzPatrick and Surry Unit 1 to permit issuing orders on August 14, 1979, and August 22, 1979, respectively allowing resumption of operation for 60 days while some remaining pipe support analyses were completed.

Surry Unit 2 was shut down for steam generator repair and replacement prior to the March 13, 1979 shut down order. Because of the long shutdown for steam generator work, the seismic reanalysis required by the order was delayed by the licensee. It is not anticipated that the required seismic reanalysis will lengthen the plant shutdown.

Several actions have been taken by the NRC staff related to review, evaluation and approval of computer codes used for seismic analysis of safety-related piping. The computer code verification program initiated by the staff has three principal parts; (1) review of actual computer code listings, (2) solution of NRC benchmark problems to compare results to known values, and (3) independent check analyses of piping problems using NRC's own computer code. Additionally the NRC staff reviewed the development of the mathematical model which represents the piping system.

On April 13, 1979, Florida Power and Light, the licensee for Turkey Point Units 3 and 4, reported that algebraic summation techniques had been utilized by Westinghouse in design of the main reactor coolant system piping. The NRC reviewed the results of Westinghouse's reanalysis, determined that the piping design was acceptable and permitted resumption of operation of both Turkey Point Units. However, as a result of this information, an NRC IE Bulletin was issued on April 14, 1979, requiring all licensees to review the computer codes used in the design of safety-related systems to determine if algebraic summation had been utilized. A total of 24 additional plants used an algebraic summation technique. Four of these plants were still under construction and had not yet been issued operating licenses. The computer codes identified were:

SHOCK II	Stone & Webster Engineering
WESTDYN	Westinghouse
DAPS	General Electric
PIPDYN II	Franklin Institute
ADLPIPE	Arthur D. Little Company

The NRC staff has required reanalysis of all affected piping, modification when necessary, and computer code verification for those codes used for reanalysis. The majority of the 20 operating reactors not designed by Stone & Webster Engineering utilized algebraic summation methods on very few piping systems and had reanalyzed these systems prior to responding to the bulletin. In a few cases (Pilgrim Unit 1, Brunswick Units 1 and 2, Indian Point Unit 3 and Salem Unit 1), the use of algebraic summation was more extensive. One unit, Salem Unit 1, has been shut down since April 1979 for refueling and other modifications, and will not resume operation until the algebraic summation issue is resolved. All other units have been resolved completely, or based upon NRC staff evaluation have been permitted to continue operation during reanalysis. In each case where continued operation was permitted (Brunswick Units 1 and 2 and Indian Point Unit 3) analysis methods utilized and the margin in the piping design to code allowable values were such that modification to piping systems was unlikely. The staff however required detailed reanalyses to confirm that the designs were acceptable.

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As described in the previous Abnormal Occurrence Report to Congress (NUREG-0090, Vol. 2, No. 1), an additional issue has been identified which can cause seismic analysis of safety-related piping systems to yield nonconservative results. The issue involves the accuracy of the information input for seismic analyses. NRC IE Bulletin 79-14 was issued on July 2, 1979 to all power reactor facilities with an operating license or a construction permit. The Bulletin, which was revised on July 18, 1979 and supplemented on August 14, 1979, directs the licensees to perform inspections of their safety-related piping systems and supports. Various categories of information were to be reported to the NRC within 30, 60, and 120 days. The NRC will then review the results and take action, as appropriate, on a case-by- case basis. Because of the conservatism and redundancy built into the piping systems, the NRC did not require the facilities to be shut down pending completion of the inspections and remedial action if required. However, one plant, Ft. St. Vrain, has shutdown pursuant to technical specification requirements resulting from nonconformances discovered during "as-built" inspections. The inspection at this plant is not complete. Currently, several significant nonconformances have been identified and are being resolved. Licensee responses for the information required within 30 days are being presently reviewed by the NRC.

Further reports will be made as appropriate.

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The following abnormal occurrence was reported first in NUREG-0090, Vol. 2, No. 1, "Report to Congress on Abnormal Occurrences: January-March 1979," and in the Federal Register (44 FR 45802) on August 3, 1979. It is further updated as follows:

79-3 <u>Nuclear Accident at Three Mile Island</u>

The Three Mile Island Unit 2 (TMI-2) plant remains in a stable condition. In late August 1979, core flow was being provided by natural circulation; core cooling was being maintained by steaming the "A" steam generator through the turbine bypass valve to the main condenser. On August 20, 1979, the hot and cold leg temperatures were 165°F and 157°F, respectively; the highest incore thermocouple indication was 253°F. Primary pressure was about 275 psig. As reported in the previous report, an alternate mode of cooling is in place in case the existing mode becomes inoperable.

The licensee (Metropolitan Edison Company) has made provision for a direct sampling capability in the containment while maintaining constant containment isolation. An existing, but unused penetration was modified to provide a sampling point by boring a hole through a blank flange. The penetration is about two feet above the water line (the water in the containment is about seven feet deep). The radiological aspects of the modifications were reviewed and approved by NRC personnel onsite. The staff has recently obtained the results of the initial containment water analysis performed by Oak Ridge National Laboratory. The three water samples analyzed were obtained on

August 25, 1979 and were taken at the water surface, approximately 4-1/2 feet below the water surface and approximately 3 inches above the containment building bottom floor. In general, the activity levels observed were less than those expected. For example, the activity levels for two of the more significant isotopes, cesium-137 and cesium-134, were 175 microcuries/ml and 40 microcuries/ml, respectively. The information will be useful in planning cleanup activities and to obtain a better estimate of core damage.

The first significant waste shipment from TMI-2 (a previous shipment consisted of three liners, shipped in mid-April to Richland, Washington) since the accident was shipped on August 7, 1979, arriving at Nuclear Engineering Company's Hanford, Washington burial facility on August 10, 1979. The shipment consisted of 157 55-gallon drums of low specific activity trash. The shipment was made without incident.

The licensee has completed work on specially-built equipment to decontaminate intermediate level radioactive waste water resulting from the accident. The water which would be decontaminated by the system (designated "EPICOR-II") is contained in tanks in the Unit 2 auxiliary building and totals approximately 265,000 gallons. The primary radioactive contaminants are iodine-131 and cesium-137 -- ranging from as much as 3 microcuries per milliliter of iodine to as much as 35 microcuries per milliliter of cesium. The NRC staff has completed an environmental assessment of the use of the "EPICOR-II" system and has issued it for public comments. The systems will not be used until authorized by the Commission.

The NRC's Office of Inspection and Enforcement (IE) issued their report of the investigation into the March 28, 1979 TMI-2 accident. The report (NUREG-0600) was issued in early August 1979. The IE investigation covered two aspects of the accident:

1. Those related operational actions by the licensee during the period from before the initiating event until approximately 8:00 p.m., March 28, when primary coolant flow was re-established by starting a reactor coolant pump, and

2. Those steps taken by the licensee to control the release of radioactive material to the offsite environs, and to implement the licensee's emergency plan during the period from the initiation of the event to midnight, March 30.

These investigation periods were selected because they include the licensee actions which most significantly affected the accident sequence and its results.

The IE investigation supported the reported population dose from the accident as developed by an ad hoc assessment group (which included representatives of various Federal agencies) and reported in NUREG-0558 which was issued May 10, 1979. The ad hoc's conclusion that the accident resulted in minimal risks of additional health effects to the offsite population was also summarized in the previous Abnormal Occurrence Report to Congress (NUREG-0090, Vol. 2, No. 1).

The IE investigation also substantiated earlier conclusions concerning the underlying causes of the accident and those factors that contributed to its severity. Inadequacies in six major areas were confirmed:

1. Equipment performance (failures and maloperation).
2. Transient and accident analyses.
3. Operator training and performance.
4. Equipment and system design.
5. Information flow, particularly during the early hours of the accident.
6. Implementation of emergency planning.

The investigation concluded that the accident could have been prevented, in spite of the listed inadequacies, if the plant systems and procedures had been permitted to function or be carried out as planned. Subsequent actions have been required by the NRC to retrain all licensed operators in an effort to preclude recurrence. Upgraded procedural instructions have also been required.

The investigation also identified up to 35 potential violations of federal procedures by the licensee. These are being further evaluated and appropriate action will be taken with the licensee.

The Lessons Learned Task Force is one of several TMI-2 related activities underway in the NRC. The Task Force was established in the NRC Office of Nuclear Reactor Regulation (NRR) to ensure the continued safe operation of licensed nuclear power plants. The purpose of the Task Force is to identify and evaluate those safety concerns originating with the TMI-2 accident that require licensing actions (beyond those already specified in IE Bulletins and Commission Orders) for presently operating reactors as well as for pending operating license and construction permit applications. The Task Force issued a status report together with short-term recommendations in late July 1979. The report (NUREG-0578) identified 23 specific requirements whose implementation was judged to provide substantial, additional protection which is required for the public health and safety. The time scale recommended for promulgation and implementation was also presented. The requirements were discussed with licensee representatives in a meeting held in Bethesda, Maryland in early August 1979. The ACRS completed its review of the Task Force short term recommendations and proposed implementation schedules during its meeting on August 9-11, 1979. In its August 13, 1979 letter to Chairman Hendrie, the Committee expressed its agreement with the Task Force recommendations, except in the case of four upon which the committee offered constructive comments to achieve the same objectives articulated by the Task Force. On September 6, 1979, the Commission held an open meeting with the staff to discuss the total set of short term recommendations, including the ACRS comments. Based on the proposed Task Force requirements, ACRS review and other comments received, the staff issued a letter on September 13, 1979 requiring that all operating reactor licensees begin implementation of the actions contained in NUREG-0578, as modified or supplemented in the letter.

The Task Force is developing longer term recommendations and plans to issue a final report in October 1979. Topics to be addressed in that report include general safety criteria, system design requirements, nuclear power plant operation, and the nuclear power plant licensing process. Additional licensing actions or requirements may be recommended by the Task Force within the next several months for backfit to operating plants and pending license applications.

A related ongoing effort in the NRC Office of Nuclear Reactor Regulation is the Bulletins and Orders (B&O) Task Force. This group is performing safety evaluations for the five Babcock & Wilcox plants shut down by confirmatory Commission Orders, and is reviewing the responses to IE Bulletins by licensees with nuclear steam supply systems designed by Westinghouse, Combustion Engineering, and General Electric. The B&O Task Force plans to publish reports that will cover the various plant designs of each of the reactor vendors noted above. The reports will deal with specific plant design aspects. Feedwater transients and small break loss-of-coolant accidents are being evaluated in considerable detail, including the review of emergency procedures and operator training for these events. These reports are scheduled to be available in the late summer of 1979.

Inspection and Enforcement (IE) Bulletin Nos. 77-05C and 79-06C was issued on July 26, 1979 to all pressurized water reactor (PWR) facilities with an operating license. The Bulletin requires that under loss of coolant symptoms, all operating reactor coolant pumps be tripped (turned off) immediately before significant voiding in the reactor coolant system occurs; certain required operator actions and analyses to be performed by the licensees were also stipulated. This revised Bulletin was issued after calculations by the PWR vendors indicated that, for a certain spectrum of small breaks in the reactor coolant system, continued operation of the reactor coolant pumps could increase the mass lost through the break and prolong or aggravate the uncovering of the reactor core.

Another related ongoing effort in the NRC Office of Nuclear Reactor Regulation is the Operating Licensing Task Force. This group reviewed the present operator licensing program and submitted reports to the Commissioners entitled, Qualifications of Reactor Operators (SECY 79-330E and SECY 79-330F). The reports contain 16 recommendations for improving the operator licensing program and provide the implementation schedules. The recommendations include modifications to training programs, more extensive use of simulators, increasing the requirements for obtaining a license, including more stringent examinations, and more NRC involvement in Requalification Programs.

As described in the previous Abnormal Occurrence Report to Congress (NUREG-0090, Vol. 2, No. 1), there are continuing investigations of the accident underway. Further actions will be considered and implemented as necessary based on the ongoing NRC staff studies, and the ongoing Presidential, Congressional, and NRC investigations. The NRC continues to have onsite staff at TMI to assure that (1) TMI-2 achieves a safe cold shutdown condition, and (2) radwaste cleanup and recovery operations are conducted in a safe manner such that occupational exposures and releases offsite are as low as reasonably achievable.

TMI-1, which was in a shutdown condition at the time of the TMI-2 accident, remains shutdown. It is estimated that it may be up to two years before TMI-1 can resume operation, considering the time necessary for licensee actions and modifications, public hearing process, and final NRC action.

Further reports will be made as appropriate.

APPENDIX C

OTHER EVENTS OF INTEREST

The following event is described below because it may possibly be perceived by the public to be of public health significance. The event did not involve a major reduction in the level of protection provided for public health or safety; therefore, it is not reportable as an abnormal occurrence.

Cracking in Main Feedwater System Piping (PWR Plants)

Description

On May 20, 1979, Indiana and Michigan Electric Company informed the NRC of cracking in two feedwater lines at the D.C. Cook Unit 2. Leaking circumferential cracks were identified in the 16-inch lines in the immediate vicinity of the steam generator nozzles. Subsequent volumetric examination (radiography) revealed crack indications at similar locations in all feedwater lines of both Units 1 and 2. As a result of a letter sent to all PWR licensees by the NRC and the issuance of Inspection and Enforcement (IE) Bulletin No.79-13, inspections are being performed at other PWR facilities. Of the 22 facilities examined to July 20, 1979, 12 have piping cracks or crack-like indications in the vicinity of the feedwater nozzles.

Presumed Cause/Mode of Failure

The mode of failure at the facilities with the most severe cracks has tentatively been identified as corrosion assisted fatigue. The cracking at these facilities has been located at a stress riser caused by machining the fitting. Internal diameter cracking of a less severe nature, which is not localized at the discontinuity, has been located at two units, Point Beach Unit 2 and San Onofre. The cracking mode at San Onofre has been identified tentatively as primarily stress assisted corrosion.

The initiating cause and driving force for the cracking has not been positively identified at this time. Factors that could contribute to the cracking include the following:

- Pipe vibrations
- Thermal stresses
- Environmental effects
- Improper pipe restraint and support
- Fabrication discontinuities

Reanalyses of normal piping system stresses and visual inspections of the feedwater lines have not, to date, uncovered any anomalies that would be expected to cause cracking. No significant deviations from proper feedwater chemistry control have been discovered. Through-wall thermal stresses due to

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alternate heating and cooling of this region have been analyzed and do not appear to be large enough to cause the degree of cracking found within the relatively short time periods of operation (approximately 1 year at Cook-2). At least three of the facilities involved have not experienced water hammer events. Thermal stresses, both high and low cycle, which could occur because of mixing of hot and cold water in the nozzle region during hot standby are also being considered, but to date have not been quantitatively analyzed pending the outcome of test programs.

Licensees for several facilities at which the cracking was most severe have agreed to install a multitude of thermocouples, strain gages and accelerometers at appropriate places on feedwater piping in the vicinity of the nozzles. During subsequent operations these instruments will be monitored in an attempt to find the cause or causes of cracking.

Safety Significance

The NRC has considered the safety significance of these cracks, and has concluded that the worst cracks found to date (Cook-2) would be unlikely to result in a significant feedwater line break (leaks are possible) in the event of an earthquake. It is conceivable, however, that a line may not survive a severe water hammer although it is unlikely that more than one line would experience a severe water hammer event simultaneously. Thus, the worst reasonable consequence would be the rupture of a single line, with which the facilities are designed to cope.

Repair Procedures

Repairs are being or have been made using somewhat improved designs for this piping region. The NRC has concluded that they are adequate pending the outcome of the test programs being conducted at several facilities. The NRC has advised other licensees to follow the conduct of these programs and that other remedial measures may be required later depending on the findings of the tests.

NRC Action

The NRC will follow closely the conduct of the several test programs to be conducted and will review and analyze the test results. The NRC will also have samples of the cracked piping analyzed metallographically. In addition, the piping designs and facility operating procedures of all PWRs are being reviewed in an attempt to find common factors which may result in cracking or preclude cracking.

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Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event which the Nuclear Regulatory Commission determines to be significant from the standpoint of public health or safety and requires a quarterly report of such events to be made to Congress. This report, the seventeenth in the series, covers the period April 1 to June 30, 1979.

During this period, there were five abnormal occurrences. Three occurred at the 70 nuclear power plants licensed to operate; one involved an indication of low water level in a boiling water reactor, the second involved damage to new fuel assemblies and the third involved deficient procedures. The other two occurred at Agreement State licensees; one involved releases of tritium and contamination of food and the second involved overexposures from a radiography source.

This report also contains information updating previously report abnormal occurrences.

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