NUREG-0090
Vol. 5, No. 3

Report to Congress on Abnormal Occurrences

July - September 1982

U.S. Nuclear Regulatory Commission

Office for Analysis and Evaluation of Operational Data

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U.S. NUCLEAR REGULATORY COMMISSION

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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 covers the period from July 1 to September 30, 1982

The report states that for this period there were two abnormal occurrences; one at the nuclear power plants licensed to operate and one at other NRC licensees. The first involved loss of auxiliary electrical power and the second involved rupture of at least one americium-241 well logging source. The Agreement States reported no abnormal occurrences to the NRC.

The report also contains information updating some previously reported abnormal occurrences.

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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 on February 24, 1977 (Vol. 42, No. 37, pages 10950-10952). 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 actions (e.g., notices of violations, 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 covers the period between July 1 to September 30, 1982.

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.

After the accident at Three Mile Island in March 1979, the NRC and other groups (a Presidential Commission, Congressional and NRC special inquiries, industry, special interests, etc.) spent substantial efforts to analyze the accident and its implications for the safety of operating reactors and to identify the changes needed to improve safety. Some deficiencies in design, operation and regulation were identified that required actions to upgrade the safety of nuclear power plants. These included modifying plant hardware, improving emergency preparedness, and increasing considerably the emphasis on human factors such as expanding the number, training, and qualifications of the reactor operating staff and upgrading plant management and technical support staffs' capabilities. In addition, each plant has installed dedicated telephone lines to the NRC for rapid communication in the event of any incident. Dedicated groups have been formed both by the NRC and by the industry for the detailed review of operating experience to help identify safety concerns early, to improve dissemination of such information, and to feed back the experience into the licensing and regulation process.

Most NRC licensee employees who work with or in the vicinity of 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.

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 by the NRC and the industry in their continuing evaluation and improvement of nuclear safety. Some of the reports describe events that have real or potential safety implications; however, 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 computer printout containing information on reportable events received from NRC licensees is routinely sent to the NRC's more than 100 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 abnormal occurrences reported by the Agreement States to the NRC are included in these quarterly reports to Congress.

FOREIGN INFORMATION

The NRC participates in an exchange of information with various foreign governments which have nuclear facilities. This foreign information is reviewed and considered in the NRC's assessment of operating experience and in its research and regulatory activities. Reference to foreign information may occasionally be made in these quarterly abnormal occurrence reports to Congress; however, only domestic abnormal occurrences are reported.

REPORT TO CONGRESS ON ABNORMAL OCCURRENCES

JULY-SEPTEMBER 1982

NUCLEAR POWER PLANTS

The NRC is reviewing events reported at the nuclear power plants licensed to operate during the third calendar quarter of 1982. As of the date of this report, the NRC had determined that the following was an abnormal occurrence.

82-5 Loss of Auxiliary Electrical Power

The following information pertaining to this event is also being reported in the Federal Register (Ref. 1). Appendix A (see general criterion 2) of this report notes that major degradation of essential safety-related equipment can be considered an abnormal occurrence.

Date and Place - On June 22, 1982, the NRC was notified by Commonwealth Edison Company (the licensee) of a sequence of events at Quad Cities Nuclear Power Station which resulted in a total unavailability of emergency diesel generator power for Unit 1 and the loss of offsite power and one emergency diesel generator for Unit 2. Quad Cities Nuclear Power Station utilizes two General Electric Company designed boiling water reactors and is located in Rock Island County, Illinois.

Nature and Probable Consequences - Diesel generators (DGs) at nuclear power plants provide emergency, onsite backup AC power in the event that normal offsite sources of AC power are unavailable. Quad Cities Units 1 and 2 have a combined total of three DGs. DG-1 is dedicated to Unit 1, DG-2 is dedicated to Unit 2, and DG-1/2 is a swing diesel that can be aligned to either unit. As a result of the sequence of events described below, normal offsite sources of AC power were available for Unit 1, but neither DG-1 nor DG-1/2 were available; simultaneously, all normal offsite sources of AC power were lost for approximately 40 minutes to Unit 2 and only DG-2 was available. For both Units, such loss of power sources can be considered a major degradation of essential safety-related equipment. The safety significance was increased by several other failures which occurred during the event, including loss of several instrumentation indications in the control room. Nevertheless, the actions taken by the plant staff were timely and attentive and Unit 2 was safely shut down. Unit 1 operation was not affected.

At the time of the event, Unit 2 was operating at approximately 95% and Unit 1 at 60% power. DG-1 was out of service for maintenance; however, DG-2 and DG-1/2 were operable. While preparing to remove the Unit 2 reserve auxiliary transformer from service for elective repairs, an equipment operator at 5:25 a.m. mistakenly pulled out the fuses for a 4-kilovolt bus instead of pulling the transformer fuses. (When the plant is producing electricity, the plant loads and instrumentation are powered by the plant's main generator via an auxiliary transformer. The reserve auxiliary transformer is available to provide offsite power when the plant is not operating.)

The operator error disconnected power to certain plant systems, including the 2B Reactor Feedwater Pump. The reduced feedwater flow caused a low water level, which automatically initiated a reactor trip. The Unit 2 main generator subsequently tripped, resulting in the loss of all normal AC power to Unit 2, as the reserve auxiliary transformer was already out of service. Both DG-2 and DG-1/2 started automatically and began to supply power to essential plant systems.

Pressure in the reactor was reduced by the automatic operation of the safety relief valve and subsequent manual actuation of power operated relief valves. In the process, one relief valve failed to open, and reactor operators actuated a second relief valve.

At 5:47 a.m., 22 minutes after the event began, the reactor operators started the 2A Residual Heat Removal (RHR) pump to begin cooling the water in the pressure suppression pool. (The pressure suppression pool is a doughnut-shaped tank surrounding the reactor containment. Water in the pool condenses the steam released by the relief and safety valves; condensing the steam transfers the heat to the water in the suppression pool.)

DG 1/2 tripped when the RHR service water pump was being started, cutting off power being supplied to various instrumentation and safety systems. The cause of the DG-1/2 trip was the actuation of underexcitation relays which protect the DG. Power continued to be supplied from DG-2.

Loss of DG-1/2 resulted in numerous alarms and loss of several control room instrument indications. In addition, the loss of DG-1/2 left Unit 1, which was still operating, without any backup source of power (since DG-1 was already out of service for maintenance) should it experience a loss of offsite power. With the loss of instrumentation, a senior operator went to the local instrument panel in the reactor building. He then established communications with the control room, and relayed information on reactor pressure, water level, and containment pressure to the reactor operators.

At 5:50 a.m., an unusual event was declared under the licensee's emergency plan, and appropriate notifications were made.

Pressure in the Unit 2 containment increased from the normal pressure of about 1.3 psig to about 4.3 psig. The principal causes of the pressure increase were due to leaking gaskets on the discharge lines of the main steam relief valves, multiple relief valve actuations to control reactor pressure, and shutdown of the drywell coolers. The latter occurred, as designed, as a result of an emergency core cooling system (ECCS) initiation signal which actuates at a drywell pressure of about 2 psig. Also as a result of this ECCS initiation signal, the high pressure coolant injection (HPCI) system started automatically and began to pump water into the reactor vessel. A core spray pump also automatically actuated but reactor pressure remained above its permissive pressure setpoint. The ECCS signal also tripped the running residual heat removal service water pump and the reactor building closed cooling water (RBCCW) pumps. A normal Group II isolation of containment occurred and functioned properly.

Licensee personnel in the meantime were restoring the reserve auxiliary transformer to service. By 6:04 a.m., 39 minutes after the event began, the transformer was operable and offsite power was restored for all affected plant systems.

Reactor pressure continued to be controlled by manual operation of the relief valves, and at 6:15 a.m. suppression pool cooling was established using the RHR system. Cold shutdown was achieved at about 5 p.m.

The plant returned to service on June 24, 1982, after appropriate maintenance and testing activities were completed.

Cause or Causes - The cause of the event can be attributed to nonconservative planning of maintenance activities, personnel error, and design error.

As stated previously, the station's auxiliary electrical power system utilizes three onsite power sources (DGs). Unit 1 and Unit 2 each has one dedicated DG and the third DG is a swing diesel that will automatically align itself to the Unit that requires it. With this arrangement, the removal of a dedicated DG from service would mean the potential unavailability of all automatic onsite emergency power sources of one Unit. The removal of the swing diesel generator causes unavailability of onsite power to one division of emergency electric power system of both Units. Because of this interdependence of onsite power sources between both Units at the station, any scheduled maintenance of the offsite power system of either Unit would affect the overall electric power system availabilities of both Units. The reserve auxiliary transformer is the primary source of offsite power for the plant. Therefore, the licensee's decision to remove the transformer from service for elective maintenance while the plant was in operation, and particularly with DG-1 already out of service for maintenance, was nonconservative (even though it was not prohibited by the plant's technical specifications).

The event was initiated by an operator error in pulling the incorrect fuse. The operator pulled the fuse for the bus rather than the transformer. This eventually led to a Unit 2 reactor scram and Unit 2 generator trip, resulting in the loss of all normal AC power to Unit 2.

Following loss of offsite power to Unit 2, DG-2 and DG-1/2 started as designed. However, later when the operator attempted to start a RHR service water pump for suppression pool cooling, DG-1/2 tripped. Succeeding attempts by the control room operator to start DG-1/2 failed. The cause of the trip was due to a design error in the DG control logic system. An underexcitation relay had been installed in 1981 in all three DGs as a modification recommended by the licensee; the relay is designed to protect the DG during testing when the DG is loaded to an energized bus and the relay protection should be automatically blocked when an auto-start signal actuates the DG. Due to a design error, this trip was unblocked when the operator initiated drywell and suppression pool cooling. The characteristic of the relay is such that it can actuate when a large motor (such as the RHR service water pump) is started. Actuation of the underexcitation relay also tripped the DG lock-out relay.

With the latter relay tripped, the DG would not restart until this relay was manually reset. Resetting of the relay was delayed since the equipment operator had been sent to the switchyard to expedite restoration of offsite power to Unit 2.

Actions Taken to Prevent Recurrence

Licensee - The licensee has taken appropriate measures to minimize the possibility for similar operator errors, including a review of procedures and additional training for operating personnel.

The DG trip mechanism, the underexcitation relay, has been removed, and the licensee is planning modifications to all diesel generators to prevent protective trips in an emergency situation.

The power operated relief valve which failed to open was replaced, and the operability of all relief valves was verified prior to the unit's returning to service. The licensee also replaced the leaking gaskets on the relief valve discharge lines which had contributed to the rise in containment pressure.

During the next refueling, the licensee plans to modify the core spray logic so that the drywell coolers and RBCCW pumps do not trip on a core spray initiation (e.g., 2 psig drywell pressure) if offsite power is available to the emergency buses.

The licensee is also considering other measures which would only allow the reserve auxiliary transformer to be removed from service for elective maintenance while the reactor is shut down with power being supplied from offsite sources through the main transformer.

The licensee submitted a special report of the event to the NRC on July 8, 1982 (Ref. 2).

NRC - The licensee had informed the NRC of the scheduled maintenance prior to June 22, 1982. NRC Region III and the NRC Senior Resident Inspector decided it was essential to have an inspector onsite throughout the scheduled maintenance due to the possible consequences associated with the removal of the reserve auxiliary transformer during unit operation. The NRC resident inspectors thoroughly reviewed the event and the followup actions taken by the licensee.

The results of the NRC inspection were forwarded to the licensee by NRC Region III on October 22, 1982 (Ref. 3). The forwarding letter expressed the NRC's concern regarding the nonconservative management decisions. Region III has recommended to the NRC Office of Nuclear Reactor Regulation (NRR) that the plant's technical specifications be modified to prohibit such nonconservative decisions. NRR is reviewing the Region's recommendation.

In addition, the NRC Office for Analysis and Evaluation of Operational Data performed an engineering evaluation of the 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 third calendar quarter of 1982. 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 third calendar quarter of 1982. As of the date of this report, the NRC had determined that the following was an abnormal occurrence.

82-6 Radiological Contamination from Well Logging Operations

The following information pertaining to this event is also being reported in the Federal Register (Ref. 4). Appendix A (see general criterion 1) of this report notes that moderate release of radioactive material licensed by or otherwise regulated by the Commission can be considered an abnormal occurrence. While no individual internal or external exposure limits were exceeded, the importance of the event was enhanced by the widespread nature of the radiological contamination (including unrestricted areas) and the significant clean up efforts and costs required.

Date and Place - On September 1, 1982, the NRC Region I office was notified by Consolidation Coal Company, Library, Pennsylvania, that one, and possibly two, americium-241 sealed sources had been ruptured during well logging operations at a field site near Jollytown, Pennsylvania.

Nature and Probable Consequences - The licensee's well logging device, used in coal exploration, included two sealed sources each containing 250 millicuries of americium-241 (a radioisotope with a 432 year half-life) as powdered oxide, compacted into a double-walled capsule. The sources are attached to a suspension cable which is lowered into a drill hole. The licensee's procedure is to lower the device (and sources) to the bottom of the drill hole, and then to withdraw the device at a controlled rate to log (profile) the hole. If the well logging device becomes wedged in the hole, the cable is designed to release, at the point of attachment to the device, when extreme tension is exerted on the cable. Recovery operations for the device can include the use of drilling to enlarge the diameter of the drill hole. The licensee had successfully retrieved wedged devices on nine previous occasions using such a procedure.

During well logging operations at a field site near Jollytown, Pennsylvania on August 19, 1982, the device became wedged at the 420 foot level in a drill hole of 950 feet total depth. While exerting considerable tension on the cable, the cable broke off about 80 feet above the device, rather than releasing at the device as designed.

During recovery operations on August 27, 1982, while drilling at a level which the licensee thought was well above the expected level of the stuck device, one (or both) of the americium-241 sealed sources was apparently ruptured by the drill bit. Apparently, the drill bit cutting through the 80 foot cable caused the device to move from the original wedged level to the drill bit. Americium-241 contamination mixed with the drilling mud used to cool and lubricate the bit. This mud was discharged to a nearby retention basin for recycling. The americium-241 contamination was not detected during licensee surveys because the survey instrument was not sufficiently sensitive for the procedures being used. Licensee representatives, believing the americium-241 sources still intact, replaced the first drilling rig with one more suited for planned recovery operations. The first drilling rig was sent to a second site nearby. On September 1, 1982, licensee representatives identified americium-241 contamination in the retention basin and immediately notified the NRC.

The immediate concerns were to determine the extent of the contamination and its concentration. The principal radioactive decay scheme of the americium series is predominantly a series of alpha particle emitters. For example, both americium-241 and its daughter neptunium-237 are alpha emitters; the latter also generates low energy x-rays. The radioactive material could be hazardous, particularly if inhaled or ingested.

Radiological surveys and contamination evaluations, both on and offsite, were performed by the NRC, the Commonwealth of Pennsylvania, and the licensee. On September 1, 1982, a Pennsylvania State inspector found contamination onsite as high as 10 millirem per hour in a small area of drilling mud. On September 2, 1982, an NRC Region I inspector found spots of contamination near the drilling rig as high as 6 millirem per hour at contact; most spots were less than 1 millirem per hour. The onsite surveys identified contamination at both drilling rigs, five vehicles, and various drilling pipes, casings and hand tools. Contamination levels ranged from 100 to greater than 1,000,000 disintegrations per minute per 100 square centimeters. The latter value is equivalent to about 0.5 microcuries per 100 square centimeters. Offsite surveys were performed at 20 private residences, a motel, and the licensee's corporate offices. These surveys identified contaminated shoes, clothing, and/or equipment at the motel, the corporate office, and nine private residences ranging from 20 to 600,000 disintegrations per minute per 100 square centimeters. Seven of the homes where contaminated articles were found belonged to work crew members, and two homes belonged to local residents who had walked onto the drilling site prior to the identification of the contamination incident. All contaminated articles were bagged and returned to the original site for storage. No skin contamination was identified on any person. The licensee is presently decontaminating equipment and the sites. The NRC guidelines for release of facilities and equipment for unrestricted use are 20 and 100 disintegrations per minute per 100 square centimeters for removable and non-removable contamination, respectively, for the radioactive material involved.

Film badges worn by licensee personnel on the sites indicated minimal exposures. Ten individuals who had the most intimate contact with the americium-241 contamination were given whole body counts and urine bioassays. No internal contamination was identified.

The consequences of this incident were that, while no individual external or internal exposure limits were exceeded, there were 11 locations where loose radioactive material was found in unrestricted areas frequented by members of the general public. The financial impact on the licensee will be substantial; clean up costs are estimated by the licensee to be as much as $1,000,000.

Cause or Causes - The direct cause of this contamination incident is the rupture of at least one of the americium-241 sources by the drill bit. The most likely cause was the presence of 80 feet of cable, still attached to the source, a recovery problem which had not been previously encountered. The drill bit cutting through the 80 foot cable apparently moved the device (from the expected, original wedged location) closer to the drill bit until the drill bit ruptured one or both sources. A contributing cause was the inadequate use of survey instrumentation which failed to identify the americium-241 contamination even though the licensee was making periodic radiological surveys.

Actions Taken to Prevent Recurrence

Licensee - The licensee, in addition to State of Pennsylvania and NRC inspectors, performed radiological surveys to determine the extent of the contamination, both on and offsite. Contaminated articles found were returned to the original site for storage. Equipment and any contaminated areas are being decontaminated. All sealed sources containing powdered americium-241 oxide used in well logging devices will be replaced with americium-241 sealed sources composed of ceramic microspheres. In addition, instrumentation will be purchased which is sensitive to low levels of radioactive contamination. Recovery procedures have been changed to eliminate drilling operations during recovery attempts.

NRC - The investigation into this incident is continuing. In addition, the NRC will review the incident to determine whether more specific information on recovery techniques will be necessary during license reviews of well logging operations.

An Inspection and Enforcement Information Notice is being considered to send to licensees to inform them of this event.

Further reports will be made as appropriate.

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 third calendar quarter of 1982, the Agreement States reported no abnormal occurrences to the NRC.

REFERENCES

1. U.S. Nuclear Regulatory Commission, "Abnormal Occurrence: "Loss of Auxiliary Electrical Power," Federal Register. (Item is being published in the Federal Register concurrently with this report.)

2. Letter from N. J. Kalivianakis, Station Superintendent, Quad Cities Nuclear Power Station, Commonwealth Edison Company, to J. G. Keppler, Regional Administrator, NRC Region III, transmitting a special report on the June 22, 1982 loss of auxiliary electrical power, Docket Nos. 50-254 and 50-265, July 8, 1982.*

3. Letter from R. L. Spessard, Director, Division of Project and Resident Programs, NRC Region III, to C. Reed, Vice President, Commonwealth Edison Company, forwarding Inspection Reports 50-254/82-10 and 50-265/82-11, Docket Nos. 50-254 and 50-265, October 22, 1982.*

4. U.S. Nuclear Regulatory Commission, "Abnormal Occurrence: Rupture of Americium-241 Source(s)," Federal Register. (Item is being published in the Federal Register concurrently with this report.)

*Available in NRC Public Document Room, 1717 H Street, NW., Washington, D.C. 20555, for inspection and copying (for a fee).

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 on February 24, 1977 (Vol. 42, No. 37, pages 10950-10952).

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 § 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 § 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 § 20 (10 CFR § 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 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 § 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 § 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 § 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 § 100 guidelines could result from a postulated transient or accident (e.g., loss of emergency core cooling system, loss of control rod system).

4. Discovery of a major condition not specifically considered in the Safety Analysis Report (SAR) or Technical Specifications that requires 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 § 100 guidelines could result from a postulated transient or accident (e.g., loss of emergency core cooling system, loss of control rod system).

For Fuel Cycle Licensees

1. A safety limit of license Technical Specifications is exceeded and a plant shutdown is required (10 CFR § 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.

APPENDIX B

UPDATE OF PREVIOUSLY REPORTED ABNORMAL OCCURRENCES

During the July through September 1982 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

75-7 Steam Generator Feedwater Flow Instability at Pressurized Water Reactors (PWRs)

The following abnormal occurrence was originally reported in NUREG-75/0090, "Report to the Congress on Abnormal Occurrences: January-June 1975," and updated in subsequent reports in this series, i.e., NUREG-0090-1; 6; Vol. 1, No. 4; and Vol. 2, No. 2. It is further updated as follows.

The latest update of this item (NUREG-0090, Vol. 2, No. 2) cited steam generator water hammer (SGWH) concerns in the feedwater lines at Zion Unit 1 and San Onofre Unit 1. Since the latest update, modifications or other provisions have been implemented at these two plants. Safety evaluation reports documenting the resolution of these concerns were published on December 12, 1979 (Ref. B-1) and April 22, 1980 (Ref. B-2) for the Zion and San Onofre facilities, respectively.

Because of the continuing occurrence of SGWHs in some Westinghouse (W) and Combustion Engineering (CE) plants, the NRC in September 1977 requested all W and CE PWR licensees to submit proposed hardware and procedural modifications necessary to prevent or mitigate SGWH. Licensee responses were subsequently evaluated by the NRC staff, and conclusions were presented in safety evaluation reports and letters to licensees. As a result of these evaluations, the NRC staff prepared a Branch Technical Position ASB 10-2, "Design Guidelines for Water Hammers in Steam Generators with Top Feedring Designs," and incorporated this position into Section 10.4.7 of NRC's Standard Review Plan, NUREG-0800 (Ref. B-3). Since Babcock and Wilcox (B&W) plants had not reported damaging SGWHs, these plants were not required to make changes. However, during April 1982 some B&W SGs were found to have damaged internal auxiliary feedring and support structures. These findings were discussed in the Second Quarter CY 1982 Abnormal Occurrence Report to Congress (NUREG-0090, Vol. 5, No. 2).

No further updates to AO 75-9 are anticipated in the NUREG-0090 series of reports. Reporting of progress regarding water hammer is provided quarterly under Task No. A-1 in NUREG-0606, "Unresolved Safety Issues Summary," Aqua Book, (Ref. B-4) and annually in the section of the NRC's Annual Report (Ref. B-5) which addresses "Unresolved Safety Issues."

This incident is closed for purposes of this report.

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78-5 Loss of Containment Integrity

The following abnormal occurrence was originally reported in NUREG-0090, Vol. 1 No. 4, "Report to Congress on Abnormal Occurrences: October-December 1978," and updated in subsequent reports in the series, i.e., Vol. 2, No. 2; and Vol. 2, No. 4. It is further updated as follows.

Review of this generic concern is continuing. The latest update of this item (NUREG-0090, Vol. 2, No. 4) stated that the NRC's "Interim Position for Operability of Containment Vent and Purge Valves" was being sent to all licensees in October 1979. All licensees' responses to the Interim Position have since been reviewed and found acceptable on an interim basis. The regional offices are monitoring licensees for continued compliance with these interim commitments.

No further updates to AO 78-5 are anticipated in the NUREG-0090 series of reports. Work in this area was transferred to multiplant action item B-24, "Venting and Purging Containment While at Full Power and Effect of LOCA"; progress on the latter is reported periodically in NUREG-0748, "Operating Reactors Licensing Actions Summary" (Ref. B-6).

The incident is closed for purposes of this report.

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79-1 Degraded Engineered Safety Features

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

As discussed in the previous update report, three safety concerns emerged from the analysis of the event that occurred at the 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.

The licensee submitted proposed corrective actions addressing these three safety concerns. The NRC has completed review and evaluation of these actions. The NRC staff's safety evaluation report was forwarded to the licensee on March 9, 1982 (Ref. B-7).

The incident is closed for purposes of this report.

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79-3 Nuclear Accident at Three Mile Island

The following abnormal occurrence was originally reported in NUREG-0090, Vol. 2, No. 1, "Report to Congress on Abnormal Occurrences: January-March 1979," and updated in subsequent reports in this series, i.e., NUREG-0090, Vol. 2, No. 2; Vol. 2, No. 3; Vol. 2, No. 4; Vol. 3, No. 1; Vol. 3, No. 2; Vol. 3, No. 3, Vol. 3, No. 4; Vol. 4 No. 1; Vol. 4, No. 2; Vol. 4, No. 3; Vol. 4, No. 4; Vol. 5, No. 1; and Vol. 5, No. 2. It is further updated as follows.

Reactor Building Entries

During the July 1, 1982 entry, preparations were made for the upcoming "Quick Look" (see below) experiment. Activities included the installation of a work platform and a trolley/hoist rig. The interior of the "B" D-ring was also decontaminated using a low-pressure water flush. During the July 8, 1982 entry, additional "D-ring" surveys were made and the nitrogen manifold for reactor coolant system (RCS) venting was installed.

Additional entries were made into the containment on July 12, 14, and 15, 1982, in support of the closed circuit television inspection of the reactor vessel internals. The RCS was vented, depressurized, and gas and water samples were taken from the reactor head vent. The water samples were found to have high turbidity and were expected to impair the camera inspection. The water sample indicated that the boron concentration was consistent with loop samples (approximately 3800 ppm), thereby indicating that the water in the reactor coolant legs is mixing with the water in the vessel.

After an entry was made on July 19, 1982 to vent and uncouple control rod 8H, the closed circuit television inspection of the reactor core (the "Quick Look" inspection) was performed during the entry of July 21, 1982. The TV camera was lowered through the center control rod leadscrew orifice. The camera field of view was limited, by water turbidity, to two or three inches. As the camera was inserted through the upper plenum of the reactor, the control rod support components could be identified, although their condition could not be ascertained. When the camera was lowered into the core region, the observers could identify rubble approximately five feet below the top of the core. No structural components could be seen above the rubble. The lateral field of view of the camera was very limited; therefore, the lateral extent of the area that was void of structural components could not be ascertained.

Except for a control rod spider assembly (which detached from the leadscrew during uncoupling), no structural components could be identified in the core rubble during the initial review of the TV pictures.

Incore thermocouple temperature indicators show that the average incore temperature had increased from approximately 102°F to 109°F (less than 2 degrees/day) since the primary water level was lowered. (Approximately 20,000 gallons of primary system water were drained to a reactor coolant bleed tank as part of the "Quick Look" inspection.)

Another entry was made on July 28, 1982, to take reactor coolant samples, bring in scaffolding, flush the vertical wall surfaces below the 305 ft elevation, take a 282 ft elevation sludge sample, and remove five snubbers.

On August 4, 1982, an entry was made to perform another camera inspection (Quick Look) of the TMI-2 core. One leadscrew on the core periphery and one leadscrew midway from the periphery and the core center were removed from the TMI-2 reactor. Initial attempts to remove a peripheral leadscrew (8P) were unsuccessful. The leadscrew could not be uncoupled from the control rod spider assembly. An alternate peripheral leadscrew (8B) on the opposite side of the core was eventually removed. Radiation levels in the proximity of both leadscrews were in excess of 50 R/hr. The maximum detected beta dose was 2000 R/hr. The radiation levels in promimity of the center leadscrew, which was removed on July 21, 1982, were less than 10 R/hr. Wet black crud was observed falling off the 8B leadscrew as the leadscrew was being removed.

The closed circuit television inspection of the core periphery was limited to two spider assemblies (top of control rod end pieces). The camera could not be maneuvered under the spider assemblies and the condition of the fuel assemblies in that region of the core could not be ascertained.

The inspection of the core between the periphery and the center (control rod 9E) revealed a rubble bed approximately five feet below the top of the core region. This was the approximate depth of the rubble bed in the center of the core. Intact pellets and pellet retaining springs were visible on top of the rubble. Individual intact non-fuel bearing rods were seen protruding from the rubble bed toward the top of the core. Some fuel assembly upper end fittings appeared attached to the underside of the plenum assembly. One end fitting was damaged. Fuel rod stubs protruded downward from some of the upper end fittings. There appeared to be some melting of structural materials in the area of the upper end fittings.

Reactor building entries were conducted on August 12 and 13, 1982 to conduct the third closed circuit television inspection of the core. The camera was inserted into the core through the leadscrew opening in control rod 9E (the same opening which was used for the core inspection on August 4, 1982). A metal rod for probing the rubble was inserted into the core through the same opening as the camera. The results of the probe inspection are not certain because the camera operator was unable to locate the probing rod with the camera. The probe operator felt the probe contact the rubble pile and observed the probe extension rods go into the reactor vessel another 14 inches while twisting the last extension rod. Based on handling the probe extension rods, the probe operator concluded that the top 14 inches of the rubble was relatively soft. Both the probe and camera operators were manipulating their equipment from the control rod drive platform, 40 feet above the rubble bed.

In addition to the core inspection, entry personnel continued work on the polar crane to assess the extent of damage. Reactor building entries were conducted on August 18 and August 20, 1982. Major activities during the entries included: water sample extraction from the reactor vessel, continued remote decontamination of the 282 ft elevation surfaces, polar crane damage assessment, and modifications to the personnel lift device (spider shafter) to extend the operating range of the lift from the polar crane to the 305 ft elevation.

Three reactor building entries were conducted on August 23, 25, and 27, 1982. During the entries, an attempt was made to uncouple the leadscrews from all the 61 control rods and the eight axial power shaping rods. The uncoupling was successful in all but three cases. Three control rod leadscrews were left coupled to their spider assemblies after repeated attempts to uncouple the bayonet type connections were unsuccessful. Prior to reactor vessel head removal, the leadscrews are normally uncoupled from the spider assemblies and raised to a parked position inside the control rod drive mechanisms. Following the uncoupling attempts, all control rod housings were left vented to the reactor building to prevent accumulation of potentially explosive gases. An attempt to inspect the reactor building below the 305 ft elevation with a closed circuit television camera was delayed due to camera transmission problems.

During the entries of August 30, September 1, September 3, September 8, and September 10, 1982, activities conducted in the reactor building included continued polar crane damage assessment, remote decontamination of the 282 ft elevation, primary coolant sampling, general housekeeping, and the installation of a manometer on the reactor vessel head to sample and measure the rate of gas generation in the reactor vessel. A closed circuit television inspection of the reactor building below the 205 ft elevation was also made.

On September 15, 1982 and September 17, 1982, portions of the reactor building dome were sprayed with a water jet (heated to 140°F) to remove loose surface contamination. Additional entry tasks included continued remote decontamination of the 282 ft elevation and general housekeeping.

A primary system gas sample was taken from the center control rod drive mechanism, indicating that the gas generated in the core was not collecting in explosive concentrations. The sample indicated that hydrogen gas was being released, but there did not appear to be any release of oxygen to support combustion. Based on the latest measurements, the gas generation rate in the reactor vessel was calculated to be less than 0.02 cubic foot per day.

Submerged Demineralization System (SDS)

The SDS processed approximately 50,000 gallons of reactor coolant system water during the third calendar quarter of 1982. In total to date, approximately 1,205,000 gallons of contaiminated water have been processed; this includes 250,000 gallons of RCS water.

Reactor Coolant System Cleanup

Except for the 50,000 gallons processed as discussed above, the cleanup of the RCS was terminated pending completion of the core "Quick Looks." Water movement causes increased turbidity and therefore hinders the ability of the remote camera to see objects farther away than a few inches.

Purification Demineralizer Inspection

On August 3 and 6, 1982, the remotely operated System In-service Inspection (SISI) robot entered the auxiliary building purification demineralizer cubicles ("A" and "B") to visually monitor conditions and retrieve information on dose fields, loose surface contamination, temperatures on vessel walls, etc. The dose fields ranged from a general area of 2-5 R/hr at the doorway entering the cubicle to 1,125 R/hr at approximately one foot from the bottom of the vessel. The cubicles were generally clean of debris and low levels of loose surface contamination were identified. Boric acid residue was noted in the cubicles near the floor drains. The temperatures on the external walls of the demineralizers were at ambient conditions (about 84°F).

The licensee is continuing preparations for measuring conditions within the purification demineralizer vessels. These two 100 ft^3 stainless steel vessels contain up to 50,000 curies each of mixed fission products deposited on organic ion-exchange resins.

EPICOR II Prefilter Shipment

On August 17, 1982, the first of 49 remaining EPICOR II Prefilters (PF-3) was shipped from TMI to the Battelle Columbus Laboratories (BCL) in West Jefferson, Ohio. This 50 cubic foot ion-exchange vessel, which was used to process accident generated water from the Unit 2 auxiliary building in 1979, contained approximately 1,800 curies of radioactive material and was shipped in a special type B cask (designed to withstand transportation accidents). The Department of Energy (DOE) took possession of this waste material at TMI and will conduct research and development at BCL. The NRC inspected the waste shipping package to ensure conformance with applicable regulations. The waste shipment arrived safely at BCL on August 18, 1982.

On August 25, 1982, the second of the 49 EPICOR II Prefilters (PF-1) was shipped from TMI to the Idaho National Engineering Laboratory (INEL) in Scoville, Idaho. The PF-1 liner and shipping cask were inerted with nitrogen as an added safety precaution to insure no combustible gases would exist during shipment. The gas composition in the liner will be maintined at less than 2.5% hydrogen and less than 0.5% oxygen. The Department of Energy (DOE) took possession of this waste onsite and will conduct research and development testing at the INEL facility.

Further reports will be made as appropriate.

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81-8 Seismic Design Errors at Diablo Canyon Nuclear Power Plant

The following abnormal occurrence was originally reported in NUREG-0090, Vol 4, No. 4, "Report to Congress on Abnormal Occurrences: October-December 1981," and updated in a subsequent report in this series, i.e., NUREG-0090, Vol. 5, No. 1. It is further updated as follows.

As discussed in the previous update report, the independent design reverification program for Diablo Canyon is being performed in two phases. Phase I involves the reverification of seismic design activities performed prior to June 1978. Phase II involves reverification of seismic design activities after June 1978 and other design activities performed by the licensee and their safety-related service type contractors.

The seismic reverification program plan, with certain modifications, was approved by the Commission on March 4, 1982. On March 19, 1982, Teledyne Engineering Services was approved as the reverification program manager. On June 18, 1982, the licensee submitted a program plan for the second phase of the reverificaton along with the proposed contractors. The second phase plan and contractors were the subjects of a Commission briefing by the staff on October 20, 1982.

Concurrent with the independent design reverification program, the licensee has contracted with the Bechtel Power Corporation to act as project completion manager for Diablo Canyon. A revised project Quality Assurance Program, reflecting the joint PG&E/Bechtel organization was approved in August 1982. The objective of the joint organization is to fulfill all requirements for reinstating the low power license for Diablo Canyon Unit 1 and for meeting all full power license requirements for both units.

The independent design verification program and the licensee/Bechtel internal technical review program have identified a number of errors and open items to date. As of September 1982 the independent program had identified 199 technical concerns requiring resolution. A number of these have subsequently been resolved and 13 have been classified as errors. These are errors in which design criteria or operating limits of safety-related equipment could have been exceeded and physical modifications, changes in operating procedures, more realistic calculations, or retesting are required to bring the plant into conformance with the original design. In addition, the licensee/Bechtel organization has identified 33 concerns within their program. Six have been resolved and 27 concerns have been classified as errors. These errors are not directly additive because there exists some overlap between the Teledyne and licensee/Bechtel errors.

In reference to the errors found to date, the licensee has stated that nothing has been found which would have prevented a system, structure or component from performing its intended safety function in the event of the postulated earthquake.

Further reports will be made as appropriate.

APPENDIX C

OTHER EVENTS OF INTEREST

The following events are described below because they may possibly be perceived by the public to be of public health significance. None of the events involved a major reduction in the level of protection provided for public health or safety; therefore, they are not reportable as abnormal occurrences.

1. Steam Extraction Line Rupture

On June 28, 1982, while operating at 95% power, Oconee Unit 2 experienced a rupture of a steam extraction line. The escaping steam caused burns to two personnel, who were hospitalized overnight and released. In addition, some nonsafety electrical equipment was destroyed. Oconee Unit 2, which is operated by Duke Power Company (the licensee), utilizes a Babcock & Wilcox-designed pressurized water reactor and is located in Oconee County, South Carolina.

The rupture occurred in the outside radius of a 375 mil thick 90° elbow where the 24" steam extraction line branched off a 42" high pressure turbine exhaust line. The rupture size was about 4 ft^2 (approximately two feet by two feet).

Upon hearing the explosion and observing an apparent loss of main steam turbine header pressure, the reactor operators suspected that a main steam line break had occurred. Nine seconds after the rupture, the reactor was manually tripped, initiating an automatic turbine trip. The failure was downstream of the main steam stop valves; thus, the turbine trip isolated steam supply to the extraction line. Systems and related parameters responded as expected following the reactor trip and subsequent recovery operations.

The steam escaping through the rupture physically destroyed a motor control center. There were, however, no safety-related loads supplied from the motor control center nor any essential loads which precluded routine plant shutdown. Steam impingement also destroyed several nonsafety-related instruments which were mounted on a panel board located six feet from the failure. Two of four turbine steam header pressure transmitters were among the instruments destroyed and were the reason for the loss of indication of steam header pressure. Safety-related steam generator header pressure instruments were not affected.

Seven minutes into the event, the unit experienced the loss of the process computer for a period of 3.5 minutes. The loss was apparently the result of a computer stall, a computer malfunction during which the computer either slows down drastically or quits. The computer was restarted with no major difficulty. The computer malfunction was later evaluated by the licensee to determine the cause and corrective actions. The reactor coolant subcooling margin monitors are supplied from the process computer and were for that 3.5 minutes inoperable. The operators ascertained subcooling during the period from reactor coolant system temperature and pressure indications which were available in the control room. Loss of the computer posed no major impedence to the safe shutdown of the plant.

The rupture was attributed to piping degradation that resulted from steam erosion. The steam erosion was accelerated by sustained reduced power operation resulting in lower quality steam in the line. Ultrasonic thickness testing performed on the elbow in March 1982 had revealed signficant erosion thinning, but that the elbow was still serviceable. At that time, the thinnest area recorded was 170 mils, micrometer readings performed after the rupture revealed a thickness of 17 mils at the edge of the failure. The inspection program performed in March 1982 may not have identified the section where the line was thinnest.

The ruptured elbow was replaced. The licensee inspected 15 extraction line fittings on Units 1 and 2. A detailed 4" x 4" grid map was marked on the areas being examined, providing for test correlation/comparison and more detailed analysis. A July 1, 1982 examination of extraction piping on Oconee Unit 1, which was operating at full power, revealed an area of approximately 4" x 4" which had been eroded from 375 mils to 100 mils, which is below the minimum wall thickness for that schedule pipe. Unit power was reduced, a patch welded on, and power returned to 100%. The licensee plans to replace the elbow during the next outage of adequate duration. The licensee also inspected main steam system piping upstream of the main steam stop valves on Oconee Unit 3. The licensee is currently re-evaluating their program of examination of the extraction lines.

Initial indication of steam extraction line degradation at the Oconee facility was discovered in 1976 when a pinhole leak occurred on a similar line in Unit 3; subsequently, an informal, undocumented maintenance surveillance program utilizing ultrasonic examination of steam extraction lines was begun. Further cases of material degradation by steam erosion have occurred at Unit 3 between 1976 and 1980; in addition, there was one case at Unit 1 in 1978. Subsequent to the 1979 event, the licensee formalized their ultrasonic thickness inspection program. The procedure required only that the measurements be performed 90° apart around the pipe but did not specify a grid map nor location of the measurements.

Inspections were performed by NRC Region II personnel. No violations or deviations were disclosed. The NRC issued Inspection and Enforcement Information Notice No. 82-22 to all nuclear power reactor licensees to inform them of this event (Ref. C-1). The Information Notice also informed the licensees of several similar failures at other facilities since January 1, 1982. All apparently resulted from steam erosion and led to plant shutdowns. These occurred at Trojan Unit 1 (January), Vermont Yankee (January), Zion Unit 1 (February), and Browns Ferry Unit 1 (June). The nuclear industry is reviewing the problem of steam erosion.

There were no radiological consequences associated with this event. The rupture did not degrade equipment required for safe shutdown of the plant. As stated previously, systems and related parameters responded as expected following the reactor trip and subsequent recovery operations. Therefore, this event is not reportable as an abnormal occurrence.

2. Degraded Safety Relief Valves

On July 3, 1982, an event occurred at Hatch Unit 1 in which reactor pressure began increasing but none of the eleven safety relief valves actuated at their prescribed setpoints. Hatch Unit 1, which is operated by Georgia Power Company (the licensee), utilizes a General Electric-designed boiling water reactor and is located in Appling County, Georgia.

Unit 1 was operating at 100% power when a spurious high-pressure signal caused a reactor scram. The main turbine had not tripped when a Group 1 isolation* occurred. High-pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) automatically started and injected; the recirculation pumps tripped. The main turbine was then manually tripped. When vessel water level recovered and reached the high water level trip setpoint, HPCI, RCIC and feed-water pump turbines tripped.

Gradual vessel repressurization continued beyond the high-pressure scram set-point (1035 psig) on a 0.5 psi/sec ramp without relief valve actuation. At about 1180 psig, three safety/relief valves (SRVs) automatically actuated, relieving vessel pressure rapidly. Upon the SRV's closure, the main steam isolation valves were manually reopened and the reactor was cooled and depres-surized to cold shutdown. During cooling and depressurizing, the remaining eight SRVs were manually actuated and functioned properly.

The SRVs installed on Hatch 1 are the two-stage Target Rock model number 7567F. All three SRVs that opened automatically were located on the same steam line and were the only valves on that line. Their setpoints were 1080, 1080, and 1090 psi. The remaining eight SRVs were set at 1080, 1090, or 1100 psi. All had been refurbished and steam set at Wyle Labs during the previous refueling outage and had most recently been actuated in August 1981.

Following the July 3, 1982 event, the top works or pilot section of all the SRVs were removed and sent to Wyle Labs, where they were tested in the as-received condition. Six passed their first test, four passed on retest, and the final valve passed on the second retest--all without setpoint spring adjust-ment. The average first actuation pressure was 0.9% above nameplate with the highest pressure required being 4.1% above nameplate. No abnormal leakage characteristics were observed for any of the valves. No apparent mechanical failure was found in the top works at Wyle Labs or the valve bodies inspected at Hatch.

Three additonal licensees--Tenessee Valley Authority, Northeast Nuclear Energy Company, and Boston Edison--had reported that two-stage Target Rock valves, tested in the as-received condition at Wyle Labs, failed to actuate within 1% of the setpoint. The Hatch 1 event was potentially the most signficant in terms of both (1) the fraction of valves that failed to open at their setpoint, and (2) the pressure above setpoint required to open the valves.

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*Closure of main steam isolation valves, main steam drain isolation valves, and recirculation loop sample isolation valves.

The General Electric Company (GE) and the Target Rock Company have joined Georgia Power in attempting to determine the cause of the failure of the valves to actuate. A GE analysis suggests that the most likely cause of the high actuation pressure is some combination of friction in the labyrinth seal area and/or sticking of the pilot disc in its seat. The slow repressurization ramp and the extended period during which the valves were not actuated are also considered possible contributors to the incident.

To define the problem and to improve the probability of actuation of the SRVs, Georgia Power has instituted a program at Hatch whereby nine of the eleven Unit 1 valves will be exercised regularly. Two valves will not be exercised and will be utilized for possible future testing. Unit 2 valves will be subjected to a similar program. Also, Georgia Power has arranged with GE and with cooperating licensees for screening tests to be done on additional SRVs at Wyle Labs. Valves which are pressurized at the 0.5 psi ramp to 103% of nameplate rating without actuating are to be candidates for diagnostic testing to determine the magnitude of forces in the disc-to-seat interface and labyrinth seal area. Further, examination of interior surfaces will be conducted to locate any physical damage. Two such candidates were found in the recent testing of three SRVs belonging to Northeast Nuclear Energy Company's Millstone Unit 1.

The NRC performed inspections at the plant. Meetings have been held with the licensee, GE, and the Target Rock Company to discuss the problem including possible corrective actions. The NRC issued Inspection and Enforcment Information Notice No. 82-41 to all nuclear power reactor licensees to inform them of this event (Ref. C-2).

Although the safety relief valves opened at a higher than expected pressure, system pressure was maintained significantly below the technical specification safety limit of 1325 psig. The event involved only a minor reduction in the degree of protection of the public health or safety. Therefore, it is not considered reportable as an abnormal occurrence.

REFERENCES
(FOR APPENDICES)

B-1 Letter from A. Schwencer, Chief, Operating Reactors Branch #1, Division of Operating Reactors, NRC Office of Nuclear Reactor Regulation, to D. L. Peoples, Director of Nuclear Licensing, Commonwealth Edison Company, transmitting a September 1979 Safety Evaluation Report on Steam Generator Water Hammer at Zion Units 1 and 2, Docket Nos. 50-295 and 50-304, December 12, 1979.*

B-2 Letter from D. L. Ziemann, Chief, Operating Reactors Branch #2, Division of Operating Reactors, NRC Office of Nuclear Reactor Regulation, to R. Dietch, Vice President, Nuclear Engineering and Operations, Southern California Edison Company, transmitting an April 1980 Safety Evaluation Report on Steam Generator Water Hammer at San Onofre Unit 1, Docket No. 50-206, April 22, 1980.*

B-3 U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition," USNRC Report NUREG-0800, published July 1981.**

B-4 U.S. Nuclear Regulatory Commission, "Unresolved Safety Issues Summary," Aqua Book, USNRC Report NUREG-0606, issued quarterly as a series.***

B-5 U.S. Nuclear Regulatory Commission, "Annual Report." These annual reports summarize the major activities of the NRC and are submitted to the President for transmittal to Congress.†

B-6 U.S. Nuclear Regulatory Commission, "Operating Reactors Licensing Actions Summary," Orange Book, USNRC Report NUREG-0748, issued periodically as a series.***

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*Available in NRC Public Document Room, 1717 H Street, NW., Washington, DC 20555 for inspection and copying (for a fee).

**Available for purchase from National Technical Information Service, Springfield, VA 22161.

***Available for purchase from NRC-GPO Sales Program, Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555 and National Technical Information Service, Springfield, VA 22161. This series of reports is also available on a subscription basis from the NRC-GPO Sales Program.

† Available for purchase from NRC-GPO Sales Program, Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

B-7 Letter from J. F. Stolz, Chief, Operating Reactors Branch #4, Division of Licensing, NRC Office of Nuclear Reactor Regulation, to W. Cavanaugh, III, Senior Vice President, Energy Supply, Arkansas Power & Light Company, Docket No. 50-313, March 9, 1982.*

C-1 U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 82-22, "Failures in Turbine Exhaust Lines," July 9, 1982.*

C-2 U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 82-41, "Failure of Safety/Relief Valves to Open at a BWR," October 22, 1982.*

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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 covers the period July 1 to September 30, 1982.

During the report period, there were two abnormal occurrences; one at the nuclear power plants licensed to operate and one at other NRC licensees. The first involved loss of auxiliary electrical power and the second involved rupture of at least one americium-241 well logging source. The Agreement States reported no abnormal occurrences to the NRC.

The report also contains information updating some previously reported abnormal occurrences.

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