NUREG-0090
Vol. 5, No. 1

Report to Congress on Abnormal Occurrences

January - March 1982

U.S. Nuclear Regulatory Commission

Office for Analysis and Evaluation of Operational Data

NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 1717 H Street, N.W. Washington, DC 20555
2. The NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, DC 20555
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the NRC/GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the *Code of Federal Regulations*, and *Nuclear Regulatory Commission Issuances*.

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal and periodical articles, and transactions. *Federal Register* notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free upon written request to the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

GPO Printed copy price: $4.75

NUREG-0090
Vol. 5, No. 1

Report to Congress on Abnormal Occurrences

January - March 1982

Date Published: August 1982

Office for Analysis and Evaluation of Operational Data
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

U. S. NUCLEAR REGULATORY COMMISSION

Previous Reports in Series
for the
Report to Congress on Abnormal Occurrences

NUREG 75/090, January-June 1975, published October 1975

NUREG-0090-1, July-September 1975, published March 1976

NUREG-0090-2, October-December 1975, published March 1976

NUREG-0090-3, January-March 1976, published July 1976

NUREG-0090-4, April-June 1976, published October 1976

NUREG-0090-5, July-September 1976, published March 1977

NUREG-0090-6, October-December 1976, published June 1977

NUREG-0090-7, January-March 1977, published June 1977

NUREG-0090-8, April-June 1977, published September 1977

NUREG-0090-9, July-September 1977, published November 1977

NUREG-0090-10, October-December 1977, published March 1978

NUREG-0090, Vol.1, No.1, Jan-March 1978, published June 1978

NUREG-0090, Vol.1, No.2, April-June 1978, published September 1978

NUREG-0090, Vol.1, No.3, July-September 1978, published December 1978

NUREG-0090, Vol.1, No.4, October-December 1978, published March 1979

NUREG-0090, Vol.2, No.1, January-March 1979, published July 1979

NUREG-0090, Vol.2, No.2, April-June 1979, published November 1979

NUREG-0090, Vol.2, No.3, July-September 1979, published February 1980

NUREG-0090, Vol.2, No.4, October-December 1979, published April 1980

NUREG-0090, Vol.3, No.1, January-March 1980, published September 1980

NUREG-0090, Vol.3, No.2, April-June 1980, published November 1980

NUREG-0090, Vol.3, No.3, July-September 1980, published February 1981

NUREG-0090, Vol.3, No.4, October-December 1980, published May 1981

NUREG-0090, Vol.4, No.1, January-March 1981, published July 1981

NUREG-0090, Vol.4, No.2, April-June 1981, published October 1981

NUREG-0090, Vol.4, No.3, July-September 1981, published January 1982

NUREG-0090, Vol.4, No.4, October-December 1981, published May 1982

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 January 1 to March 31, 1982.

The report states that there were four abnormal occurrences at the nuclear power plants licensed to operate. The first involved diesel generator engine cooling system failures. The second involved pressure transients during shutdown. The third involved major deficiencies in management controls. The fourth involved a steam generator tube rupture. There were no abnormal occurrences for the other NRC licensees during the report period. The Agreement States reported no abnormal occurrences to the NRC.

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

CONTENTS

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 January 1 to March 31, 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.

REPORT TO CONGRESS ON ABNORMAL OCCURRENCES

JANUARY-MARCH 1982

NUCLEAR POWER PLANTS

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

82-1 Diesel Generator Engine Cooling System Failures

Preliminary information pertaining to the event was reported in the Federal Register (Ref. 1). Appendix A (one of the general criteria) of this report notes that major degradation of essential safety-related equipment can be considered an abnormal occurrence.

Date and Place - On October 23, 1981, the NRC was notified by the Commonwealth Edison Company (the licensee) concerning trips (shutdowns) experienced by the Dresden Station's Unit 2/3 and Unit 3 diesel generators. The cause was attributed to insufficient cooling water flow to the diesel generator heat exchangers resulting in high engine temperature protective trips. On November 19, 1981 a similar event occurred with Unit 3 diesel generator. On December 1, 1981 the Unit 2/3 diesel generator was declared inoperable when a problem developed with Unit 2/3 diesel generator cooling water pump motor bearings. Dresden Units 2 and 3 utilize boiling water reactors and are located in Grundy County, Illnois.

Nature and Probable Consequences - Diesel generators (D/Gs) at nuclear power plants provide emergency, on-site backup AC power in the event that normal off-site sources of AC power are unavailable. Dresden Units 2 and 3 have a total of three D/Gs. "Unit 2 D/G" is dedicated to Dresden Unit 2, "Unit 3 D/G" is dedicated to Dresden Unit 3, and "Unit 2/3 D/G" is shared by Dresden Unit 2 and Unit 3. For the events described below, normal offsite sources of AC power remained available; however, the loss of both D/Gs to either plant can be considered a serious reduction in safety redundancy. For the events described below, the safety significance was enhanced since there was the potential for the simultaneous loss of all three D/Gs. The D/Gs are designed to power certain essential safety-related equipment in the event that all off-site AC power is lost.

Starting October 23, 1981, a series of inoperable D/G events occurred due to insufficient cooling water flow to the D/G heat exchangers. Subsequent investigations eventually led to the discovery of broken check valves in the discharge of the D/G Cooling Water Pump (DGCWP) for all three diesel generators (see Figure 1). The valves are horizontally mounted Crane, 8-inch, tilting disk check valves, Type 373, and have a pressure rating of 125 psi. For the last event, an indicated pressure drop was actually caused by DGCWP bearing wear; subsequent examination showed that the check valve was broken, but apparently had not caused restriction at this time. The licensee's discovery of the failed

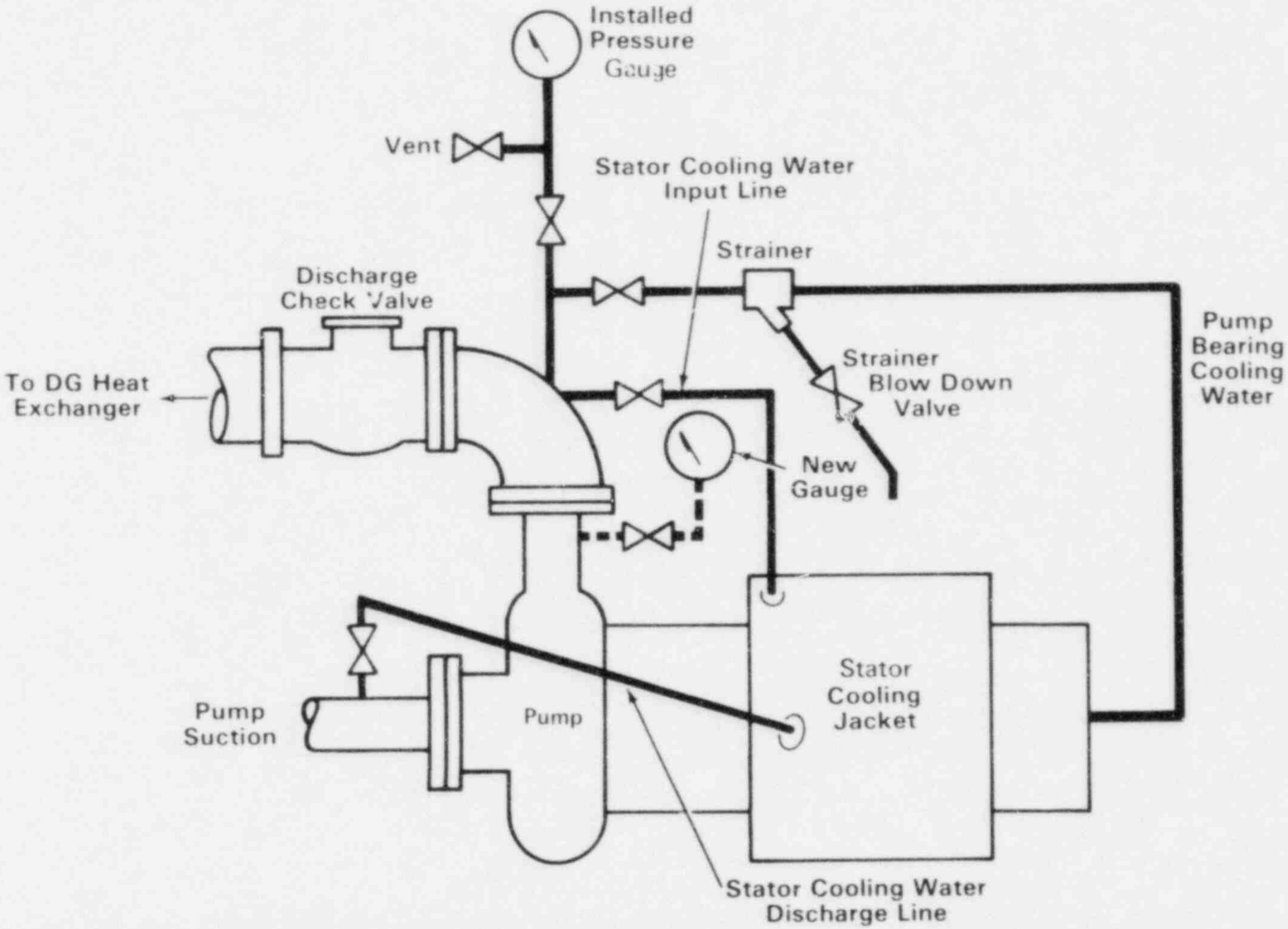

Figure 1. Diesel Generator Cooling Water Pump (DGCWP) and Discharge Check Valve

check valves which were involved in degraded flow was seriously hampered by the lack of adequate installed pressure and flow monitoring instrumentation. In the earlier stages of the licensee's investigations, the licensee suspected such possible causes for the degraded flow as air binding in the pumps, pump runout, or blockage of the suction piping to the D/G cooling water pumps. The various events are described by date and involved D/Gs as follows:

October 23, 1981 Events (Unit 2/3 and Unit 3 D/Gs)

The first loss of D/G cooling water flow was experienced October 23, 1981 while conducting a monthly D/G surveillance test. One D/G was started in the normal manner from the control room while the equipment operators were in the Unit 2/3 D/G room. After taking initial log readings, the operators left the room. At approximately 0224, the Unit 2/3 D/G tripped on high engine temperature. The D/G cooling water pump tripped when the D/G tripped as per design. The operators returned to the room, followed by two shift foremen. The records show that the Unit 2/3 pump had started at about 0228. The pump was manually cycled several times to determine if the problem was in the pump. Cooling water pressure at the D/G heat exchanger remained abnormally low. The pump was then shut down.

The two shift foremen proceeded to the cribhouse to examine the pump. The foremen, suspecting air binding as a possible cause of the problem, directed the cycling of the pump during which time the pump was vented. However, no air was observed issuing from either of two vent paths which shared a common 1/2-inch tap with the pump discharge pressure gauge and motor bearing cooling water line. The foremen observed low pump discharge pressure, low vent flow, abnormally low levels of pump noise and vibrations, and abnormally warm stator cooling line. The foremen also reported that valve positions were visually checked and found to be proper. The stator cooling line is supplied by a separate tap on the pump discharge from the tap for the vent, discharge pressure gauge, and stator cooling water lines. The low pump discharge pressure was later determined to be the result of venting through the same tap as the pressure gauge.

Since the Unit 2/3 pump can be supplied power from either Unit, the power was switched from a Unit 2 source to a Unit 3 source to determine if the low cooling water flow was caused by an electrical problem. The pump was cycled again, but the heat exchanger pressure remained low. The power supply was returned to normal, and the Unit 2/3 pump was declared inoperable.

At approximately 0400, the same monthly surveillance test was commenced on the Unit 3 D/G. Indications of insufficient cooling water flow were observed in the D/G room and reported by the equipment operators, and a D/G shutdown was commenced by the control room operators at 0407. A few seconds later, the diesel tripped on high engine temperature. The Unit 3 D/G cooling water pump tripped with the diesel as per design. The Unit 3 pump was cycled several times. During this period, the pump was vented. No air was observed during venting, and the pump performance was essentially the same as for the Unit 2/3 pump. Valve line ups again were checked and found to be proper.

At about 0417, immediately after one of the pump starts, the shift foremen and the shift engineer heard considerable noise from the Unit 3 pump and observed an increase in the pump discharge pressure. In the D/G room, equipment operators observed the D/G heat exchanger pressure return to normal. A hot restart of the Unit 3 D/G was conducted at which time the pump and D/G functioned normally. Subsequently, the Unit 2/3 pump was tested, found to operate satisfactorily, and was returned to service.

November 19, 1981 Event (Unit 3 D/G)

On November 19, 1981 at approximately 0453 during the conduct of a surveillance test of Unit 3 D/G, the diesel tripped on high engine temperature. Still suspecting air binding, the licensee cycled the cooling water pump twice, during which time the pump was vented. The individual who vented the pump observed what appeared to be an air and water mixture issuing from the vent path for about five minutes.

The Unit 3 pump was declared inoperable and the Unit 3 D/G was removed from service. Dresden Unit 3 then operated under a Technical Specification limiting condition for operation while the event was further investigated.

A broken check valve on the discharge of the Unit 3 pump was found and replaced. The valve disk had broken free of the pivot arm and was lodged in the discharge side of the valve, restricting nearly all flow.

The licensee subsequently stated that the apparent air/water mixture which was observed to have issued from the discharge pressure gauge test connection valve of the Unit 3 pump was due to the atomizing effect that occurred by partially opening the valve (gate valve). This effect was also observed by NRC inspectors.

December 1, 1981 Event (Unit 2/3 D/G)

On December 1, 1981, the Unit 2/3 pump exhibited a slow decrease in indicated discharge pressure accompanied by increasing noise and vibration levels. This decrease in indicated pressure and the increase in noise and vibration levels were later determined, through visual inspection, testing, and determination of actual bearing clearance to be caused by excessive wear on the pump motor bearings. Since the discharge pressure gauge and the motor bearing cooling water supply share a common tap on the pump discharge, the gauge responded to changes in bearing cooling water flow. This same tap also provides the two vent paths for the pump: the discharge pressure gauge test connection and the blowdown line for the Y-strainer in line with the bearing cooling water supply.

It is believed that bearing wear may have been accelerated by frequent venting of the pump while it was running, as required by Confirmation of Action Letters issued by the NRC on November 20 and 25, 1981. The letters were issued in order to provide adequate assurance that on-site emergency power would be available in the event of an accident, while the licensee continued to investigate the cause of the D/G cooling water insufficiencies. The frequent venting was consistent with the licensee's belief, at the time, that a likely cause of the flow restrictions was air binding in the D/G cooling water pumps.

The Unit 2/3 pump was replaced. During the pump replacement, the licensee inspected the pump's check valve and found it to be broken. As was the case with the Unit 3 pump discharge check valve, the disk had broken free of the pivot arm. In the case of the Unit 2/3 pump, however, the disk had not lodged into the body of the valve, but was free to move in any direction within the valve body. The valve was replaced.

During the October 23 through December 1, 1981 events, the Unit 2 D/G had functioned properly. However, as part of the investigation, the Unit 2 pump discharge check valve was inspected. It was found that this valve was also broken, but at a different point. The valve hinge broke and the pivot hinge remained rigidly attached to the valve disc. This valve was also replaced.

These events were unique insofar as D/G failures are concerned because all three check valves were found to be broken during a short period of time, diagnosis of the valve failures was delayed due to inadequate and poorly designed instrumentation, both D/Gs of Dresden Unit 3 were simultaneously affected (and made inoperable) on October 23, 1981, and the potential existed for all three D/Gs to be affected simultaneously had the Unit 2 D/G check valve broken in the same manner as the other two check valves.

Cause or Causes - The D/Gs were made inoperable due to insufficient cooling water flow to the D/G heat exchangers. In most of the events, the degraded flow was caused by broken check valves in the cooling water pump discharge. In one event, even though the check valve was broken, it was not restricting flow; a decrease in discharge pressure was caused by worn bearings on the D/G cooling water pump. For the check valves for Unit 3 and Unit 2/3 D/Gs, the valve disk had broken free of the pivot arm. For Unit 2 D/G, the pivot arm remained attached to the valve disk, but was broken at the hinge to the valve body. The disks for the Unit 3 and Unit 2/3 check valves were at times in such a position that flow was restricted to the cooling water heat exchanger which caused the D/Gs to overheat. The disk for the Unit 2 check valve had apparently not located itself in a critical position to affect flow.

It is not known how long these check valves were broken before they were detected since the broken Unit 3 and Unit 2/3 check valve discs were free to move within the valve bodies and may have been that way for some time before coming to rest in a position which would restrict flow enough to cause the D/G to trip on high engine temperature.

The check valves are not routinely covered by inservice testing programs or routine surveillance to verify valve operability. These failures were not adequately characterized by operator observations and available instrument readings during diesel generator surveillance tests, but were discovered by direct inspection of the internals of the check valve. The use of a common tap for pump discharge pressure, pump bearing cooling and pump discharge line venting, and the lack of alternate methods to adequately monitor flow actually delayed and confused the analysis of the problem.

Actions Taken to Prevent Recurrence

Licensee - Commonwealth Edison conducted an investigation of the events and plans to take or has taken the following specific actions:

1. All three discharge check valves have been replaced.

2. Instrumentation changes for the cooling water systems for all three diesels will be made. An additional pressure gauge on the discharge volute of each DGCWP has been installed (see "New Gauge" in Figure 1). When an engineering study is completed, the licensee will also provide a more accurate indication of system flow than presently exists (i.e., 0-200 psig gauges on the inlet and outlet of the heat exchangers are unreliable to indicate a small differential pressure).

3. Plant procedures will be changed to lower the probability of air leakage into the pumps or inadvertent shutting of the pump suction valves. In addition, motor bearing tolerances for the pumps, which are checked annually, will be recorded for trend analysis purposes.

4. The electrical supply and control systems have been extensively tested in an effort to determine that the DGCWPs operated properly. No negative results were found; however, the pump motor electrical overload devices were changed so that they will reset automatically instead of manually.

5. Because each diesel experienced a defective check valve, the licensee plans to examine and test each valve annually.

The following testing was also conducted to determine the cause of the events:

1. Systems tests to verify or deny the possibility of air binding or suction blockage. It was concluded that these were not contributing factors to the events.

2. Tests to determine normal indications.

3. A test to verify that pump runout did not occur.

4. Radiography of valves in the cooling water system.

Pump operability tests were conducted periodically during the course of the events as needed to provide additional assurance that the systems would operate properly if called upon.

NRC - An inspection team was dispatched to the site on October 23, 1981, to begin to determine the adequacy of the licensee's response to the initial event.

Due to the licensee's inability to identify the cause of the event to the inspector's satisfaction, the licensee agreed to run surveillance on the pumps daily to verify operability. This team returned to the site on October 27, 1981, to gather additional information. In a telephone call between the Plant Superintendent and NRC Region III Staff on October 28, 1981, the licensee was requested to increase the intensity of their investigation into the event. On October 30, 1981, NRC Region III management was briefed by the inspectors. Since the initial data had a number of inconsistencies, it was decided that an investigator should be assigned and that sworn statements would be taken to ensure that the information received had a greater level of reliability. During the next four weeks, seven site visits were made during which eight sworn statements were taken.

The NRC issued a series of Confirmation of Action Letters to the licensee in order to provide adequate assurance that onsite emergency power would be available in the event of an accident, while the licensee continued to investigate the cause of the D/G cooling water insufficiencies. The first Confirmation of Action Letter was issued November 20, 1981 (Ref. 2). As more information became available and was analyzed, the Confirmation of Action Letter was superseded by a letter dated November 25, 1981 (Ref. 3); the latter was in turn superseded by a letter dated December 2, 1981 (Ref. 4).

After the licensee's investigations and corrective actions were considered adequate, the licensee was verbally released from the requirements of the December 2, 1981 letter on December 24, 1981.

On January 22, 1982, the NRC Region III forwarded their inspection report of the events to the licensee (Ref. 5). No items of noncompliance with NRC requirements were identified during the course of the inspection.

On March 26, 1982, the NRC issued IE Information Notice No. 82-08 ("Check Valve Failures on Diesel Generator Engine Cooling System") to all nuclear power reactor facilities holding an operating license or construction permit to inform them of the event (Ref. 6). Recipients are expected to review the Information Notice for possible applicability to their facilities. This Information Notice represents the generic action taken by the staff thus far.

This incident is closed for purposes of this report.

* * * * * * *

82-2 <u>Pressure Transients During Shutdown at a Nuclear Power Plant</u>

Preliminary information pertaining to this event was reported in the Federal Register (Ref. 7). Appendix A (Example 12 of "For All Licensees") of this report notes that recurring incidents which create a major safety concern can be considered an abnormal occurrence.

Date and Place - The licensee, Florida Power and Light Company, reported that on November 28 and 29, 1981 two reactor coolant system pressure transients occurred while the Turkey Point Unit 4 was shutdown. Unit 4 is a Westinghouse designed pressurized water reactor facility located in Dade County, Florida.

Nature and Probable Consequences - In 1976, the NRC noted an increasing number of incidents called "pressure transients" that were occurring in pressurized water reactors (Ref. 8). The term "pressure transients," as used here, refers to incidents where the temperature-pressure limits of the reactor vessel (included in the facilities' Technical Specifications) were exceeded. The majority of the incidents occurred during startup or shutdown operation when the reactor coolant system was at low temperature. About 30 incidents had occurred; eight occurred in 1976. Concern existed for the possibility of a reactor vessel failing by the brittle fracture mechanism as a consequence of a pressure transient at near ambient temperature (near 100°F), once the reactor vessel material has experienced a reduction in fracture toughness (an upward shift in nil-ductility transition temperature) due to irradiation effects which gradually accumulate over an extended period of time. In order for a reactor vessel to fail, in addition to the low temperature, high pressure and loss of fracture toughness conditions, it must also have a critical-sized flaw at a high stress location in the embrittled area, i.e., that part of the cylindrical shell of the reactor vessel directly opposite the core (the belt line area). In 1976 many reactor vessels had not yet experienced a significant reduction in fracture toughness and conservatism existed in reactor vessel design and fabrication control to preclude sizeable flaws. However, because of the potential safety significance of such incidents occurring when the reactor vessels became more embrittled, the NRC requested the licensees to upgrade administrative controls in the near term to reduce the likelihood of future pressure transients and to install design modifications by the end of 1977 (Ref. 9) to further reduce their likelihood of occurrence and mitigate their consequences.

The pressure transients, described below, that occurred at Turkey Point Unit 4, exceeded by a factor of two the temperature-pressure limits stated in the Technical Specifications which are based on Appendix G of 10 CFR Part 50 (which relates to Section III of the ASME Boiler and Pressure Vessel Code). Fracture mechanics analysis indicated, however, that there was no significant impairment of the reactor vessel integrity. Concerns existed because Turkey Point Unit 4 has a reactor vessel with sufficient radiation exposure to reduce the fracture toughness of the reactor vessel at low temperatures, and the pressure transients had the potential for brittle fracture of the reactor vessel if significant flaws were present and the transients had not been promptly terminated by operator action. These transients highlight the importance of properly operating over-pressure mitigation systems to reduce the potential for brittle fracture of the reactor vessel. Though the frequency of pressure transients has decreased, the possibility of affecting a reactor vessel's integrity remains as a safety concern. Any event which impacts on the integrity of the reactor vessel is a significant safety matter and would likely require significant actions such as an inservice inspection prior to further operation with additional surveillance, repair, and annealing of the vessel, as necessary.

Conditions Prior to the Pressure Transients

The reactor was shutdown and preparations were underway to restart from a refueling outage. The plant operators were performing OP 0202.1 - Reactor Startup - Cold Condition to Hot Shutdown Conditions. The Reactor Coolant System (RCS) had been filled solid with water. The letdown path was via the Residual Heat Removal (RHR) system suction valves MOV-4-750 and 751, which close at 465 psig to prevent overpressurizing the RHR system. The RHR system was cross-connected to the letdown portion of the Chemical and Volume Control System (CVCS) downstream of the RHR heat exchangers at valve HCV-4-142. Letdown flow control to the Volume Control Tank and consequently, RCS pressure, was controlled by pressure control valve PCV-4-145 in the letdown portion of the CVCS. One of three positive displacement charging pumps was in operation providing both makeup into the RCS and Reactor Coolant Pump seal injection flow. RCS temperature was about 110°F and pressure was about 340 psig.

With the plant alignment described above, any flow blockage in the letdown path would cause an immediate increase in RCS pressure because the charging pump would be charging into a water solid system. Overpressure mitigating devices installed include an alarm at 400 psig warning of impending overpressure mitigating system (OMS) protective action and two independent OMS channels designed to both alarm and operate power operated relief valves (PORVs) on the pressurizer at 415 psig (at low temperature) and prevent an unacceptable pressure excursion. Figure 2 shows a schematic of the Turkey Point Unit 4 pressure control system, together with a schematic of the overpressure mitigating system.

At the time of the incidents, however, one OMS train was known to be inoperable, i.e., the PORV block valve was shut. Maintenance was being conducted on the high pressure controls for the PORV of that train. Unknown at the time, a blown fuse in the OMS comparator output rendered inoperable the alarm that signals a need for primary OMS protective action at 415 psig. Also unknown at the time, the backup OMS train was inoperable because (1) the root isolation valve for its pressure transmitter, PT-4-405, was shut which rendered the system inoperable during the first event, and (2) the temperature summator for the train had failed high rendering the train inoperable.

Description of the Pressure Transients

On November 28, 1981, at 10:55 p.m., the 4B Reactor Coolant Pump (RCP) was started to begin RCS heatup. The Reactor Control Operator noticed that RCS pressure was approximately 500 psig and increasing. Though it is common for the RCS pressure to surge momentarily following RCP startup, the operator noted that conditions persisted and were thus abnormal. He also noticed that valve PCV-4-145 was in the fully closed position and attempted to open it automatically by lowering the control setpoint. When this attempt failed, the valve was opened using the manual control mode, and the 4B RCP, 4A charging pump, and the pressurizer control heaters were shut off. One Power Operated Relief Valve (PORV-4-455C) was opened by the operator to reduce RCS pressure. The other PORV (PORV-4-456) was isolated and out-of-service for maintenance on the high pressure controls for the PORV. An RHR isolation valve (MOV-4-750) was

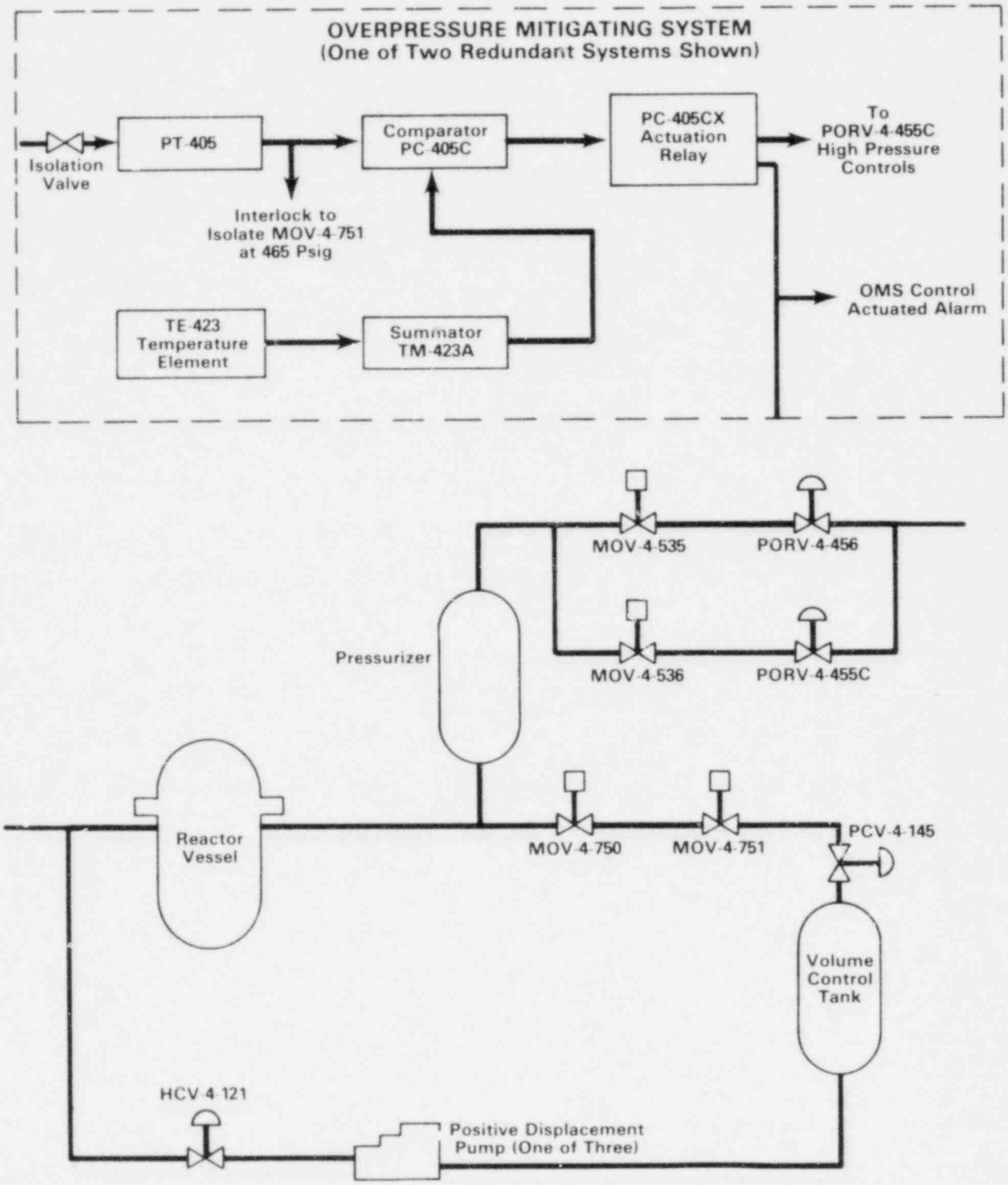

Figure 2. Turkey Point Unit 4 - Schematic of Pressure Control System

found in the closed position and was immediately opened by the operator. PCV-4-145 was returned to auto-control and the 4A charging pump was restarted. The RCS pressure was then maintained constant at approximately 335 psig.

The RCS peak pressure during the transient was 1100 psig. Duration of the overpressure condition was approximately two minutes.

The pressure transient was initially diagnosed as initiating from misoperation of valve PCV-4-145. The root isolation valve for PT-4-405 was also found closed which made the backup OMS train inoperable. The root isolation valve was opened; valve PCV-4-145 was returned to auto-control and RCS pressure was maintained constant.

On November 29, 1981, at 12:55 a.m., the 4B RCP was restarted. An overpressure condition recurred with peak pressure reaching 750 psig. Again the RCP and the charging pump in operation were shut down. PORV-4-455C was manually opened to decrease RCS pressure. Duration of the overpressure condition was approximately one minute.

During both occurrences, the operators took action to stop the charging pumps which were providing the source of rapid pressurization. However, once the letdown flow was significantly reduced or terminated by closure of the RHR system isolation valves, timely operator action would be ineffective because of the rapidity of the transient.

Cause or Causes - A pressure increase occurred when starting the RCP which exceeded the magnitude expected for a normal RCP start. Contributing to the pressure transients were the subsequent automatic closure of the RHR system suction isolation valves and the malfunction of the OMS while operating in a water solid condition. The automatic closures of the RHR system suction isolation valves were attributed to:

(1) RCS pressure transmitter PT-4-403 sensing a high pressure and closing MOV-4-750, due to the pressure interlock at 465 psig during the first event, thus resulting in the charging pump operation overpressurizing the system.

(2) PT-4-405 (the backup OMS input) was reading about 130 psig higher (based on post event testing) than actual RCS pressure when unisolated after the first event. (The transmitter had been relocated and its setpoint may have changed due to hydrostatic testing of the transmitter together with its sensing line.) This variance likely led to MOV-4-751 closing at about 375 psig actual RCS pressure, initiating the second pressure transient.

The reasons the OMS did not operate as designed are:

(1) One train was inoperable for maintenance as permitted by license conditions.

(2) The backup train failure was attributed to:

(a) The root isolation valve to PT-4-405 was shut, isolating PT-4-405, during the first event. (No procedure was found that aligns RCS instrumentation root valves prior to RCS fill.)

(b) In addition, during both events, the backup OMS temperature summator, which generates the "pressure set point" to which loop pressure is compared to generate the OMS actuation signal, had failed high - about 2335 psig - also rendering the backup OMS inoperable. This condition was unknown because of an inadequate surveil'ance procedure used to satisfy the technical specification requirement to operationally check each channel. The procedure is OP 1004.4 - Overpressure Mitigating System Functional Test of Nitrogen Backup System - dated May 7, 1981. This procedure did not test the summator.

Actions Taken to Prevent Recurrence

Licensee - After the first pressure transient, the root valve to PT-4-405 was reopened. In addition, attempts were made to release the redundant OMS loop from clearance and restore it to operating condition, but this was not accomplished by the time the second pressure transient occurred. The immediate corrective action during both events consisted of reducing the RCS pressure to a value within the Technical Specification limits. Subsequent to the second event, the licensee requested an evaluation of the consequences from the Nuclear Steam System Supplier (Westinghouse) and notified the NRC's Region II of the incidents. The licensee also confirmed that the Unit would not be restarted until the NRC has reviewed the results of the requested analyses.

A fracture mechanics analysis based on the methods of Appendix G to Section III of the ASME Boiler and Pressure Vessel Code was performed by Westinghouse. The analysis showed that the integrity of the reactor vessel was not impaired by these transients. It was further judged that the fatigue life of the vessel was not significantly affected. An independent licensee consultant reviewed the analysis and concurred with its conclusions. The fact that there was no thermal stress present was a beneficial factor in the analysis.

The licensee responded to the NRC's notice of violation by taking appropriate actions. Procedure changes were made to include additional equipment checks as well as to insure proper valve line up following any tests prior to releasing the systems to operations. These actions will minimize the probability of component failures similar to the ones that resulted in the OMS operational anomalies.

NRC - The NRC conducted a special safety inspection of the circumstances related to these events (Ref. 10). The NRC's Region II reviewed the analysis of the consequences of the events prior to the unit returning to operation. The licensee was cited with a notice of violation for (1) having an inadequate functional testing procedure for the OMS in that the summator circuitry was not tested, and (2) not including an alignment check of the instrumentation root valves in station procedures for reactor coolant system fill after refueling or plant startup.

NRC Inspection and Enforcement Information Notice No. 82-17 ("Overpressurization of Reactor Coolant System") was issued to other licensees informing them of these events and their potential significance (Ref. 11).

This incident is closed for purposes of this report.

* * * * * * *

82-3 Major Deficiencies in Management Controls at a Nuclear Power Plant

Preliminary information pertaining to this event was reported in the Federal Register (Ref. 12). Appendix A (Example 11 of "For All Licensees") of this report notes that serious deficiency in management or procedural controls in major areas can be considered an abnormal occurrence.

Date and Place - On January 18, 1982, the NRC issued a Notice of Violation and Proposed Imposition of Civil Penalties for $550,000, together with an Order Modifying the License Effective Immediately, to Boston Edison Company (licensee for the Pilgrim Nuclear Power Station) for management control deficiencies, involving control of combustible gases inside containment and maintenance activities pertaining to the reactor core isolation cooling system. On February 4, 1982, the licensee was further cited for various violations, including inadequate management controls for operation of the plant with drywell temperatures in excess of design values. The Pilgrim Nuclear Power Station (Pilgrim Unit 1) utilizes a boiling water reactor and is located in Plymouth County, Massachusetts.

Nature and Probable Consequences - Three occurrences of safety significance at Pilgrim Unit 1 indicated continuing serious deficiencies in management control of certain licensee activities. Two of the occurrences involved time periods of several years. Although different in nature, these three occurrences demonstrated a recurring lack of management attention to those licensee activities important to safety. Two of these items were examined by NRC inspections conducted during the period June 15 to September 30, 1981 (Ref. 13). The third item was identified during NRC inspections conducted between November 1, 1981 and January 18, 1982 (Ref. 14).

The first item involved failure of the licensee from November 27, 1978 to June 5, 1981 to comply with the provisions of 10 CFR 50.44 regarding the ability to control combustible gas mixtures following postulated accidents. The safety significance of this item is that the ignition of an uncontrolled accumulation of combustible gases inside containment during certain postulated accident conditions could have resulted in deflagration and a pressure surge of the containment atmosphere having the potential to breach the containment and to release substantial quantities of radioactive material to the environment.

The second item concerned violation of a Technical Specification limiting condition for operation. NRC inspections showed that improper management control of maintenance activities on safety-related electrical power supplies resulted in a degradation of the containment automatic isolation control logic, partially disabling two isolation valves, and thereby significantly reducing the assurance that the valves would automatically close when required. The safety significance associated with this item is that failure of these valves to close when required during certain accident sequences would cause a significant increase in the amount of radioactive materials released to the environment.

The third item involved operation of the facility at various times from plant startup in 1972 until September 26, 1981 with the primary containment drywell temperature greater than stipulated in the Final Safety Analysis Report. Not only had the licensee been aware of the situation for several years, but there was no evidence that safety evaluations had been made as required by 10 CFR 50.59. The safety significance of this item is that operation at the elevated temperatures for sustained periods could result in detrimental effects (e.g., premature aging) to equipment required to safely shut down the reactor and to mitigate certain postulated accidents.

Further details of these items are discussed below.

In regard to the first item, on May 29, 1981 the NRC was notified by the licensee that Pilgrim Unit 1 was not in compliance with the provisions of 10 CFR 50.44 pertaining to the control of post accident combustible gas mixtures in containment. On November 27, 1978, 10 CFR 50.44 became effective and required that licensees of light water reactors conduct analyses regarding hydrogen evolution following certain postulated accidents and make appropriate design and equipment modifications such that the combustible gases would be controlled. Pilgrim Unit 1 was required as a minimum to have a system capable of purging the containment to the atmosphere following a postulated loss of coolant accident. The purging system was required to meet certain design criteria which included equipment redundancy to assure system function in the event of single component failure or loss of offsite power. However, the facility operated from November 27, 1978 until June 5, 1981 with a containment atmosphere combustible gas control system which did not meet all the requirements of 10 CFR 50.44.

Associated with this violation was a material false statement, described below, involving the licensee's statement of compliance with the NRC regulations and subsequent failure of the licensee to notify the NRC of deficiencies after the licensee became aware of them.

The second item concerned operation of the facility in violation of a Technical Specification limiting condition for operation for primary containment integrity. On September 12, 1981, during electrical maintenance activities, operating personnel de-energized electrical power supplies, which partly disabled the automatic isolation control logic electrical circuits for both of the redundant containment isolation valves in the reactor steam supply pipe to the reactor core isolation cooling system. This resulted in a loss of redundancy provided in the design of the electrical circuits to assure automatic closure of these valves during certain postulated accidents. The facility was operated in this condition until September 16, 1981 (for a total of about 89 hours) when the misoperation was discovered by the NRC Resident Inspector.

The third item concerned operation of the unit at various times between plant startup in 1972 until September 26, 1981 with the primary containment drywell temperature greater than the Final Safety Analysis Report (FSAR) specified value of 150°F. The FSAR also specifies that the primary cooling and ventilation system be designed to maintain containment temperature at an average value of 135°F (148°F following a reactor scram). Even though the licensee had been aware of the situation for several years, there was no evidence that a safety evaluation had been made as required by 10 CFR 50.59. This apparent lack of management attention to high drywell temperature was probably the root cause of an incident involving a malfunction of instrumentation important to safety which occurred on September 26, 1981. During a routine reactor shutdown and cooldown for refueling, level oscillations of reactor water level instruments were observed. These oscillations occurred four times at approximately 20 minute intervals. Each of these instrument oscillations resulted in a high level automatic isolation of turbines followed by a low level automatic reactor scram and primary containment isolation.

Following the initial oscillation, the operators conducted an isolation verification, a check of redundant level indication and a survey to determine any loss of coolant inventory. A check was also made of the drywell and coolant temperatures. The 0-400" shutdown wide range level instrument showed no oscillation and the survey produced no indication of any loss of coolant from the reactor. The drywell temperature at the higest elevation was 240°F and the coolant temperature was 220°F. It was concluded that the actual reactor water level was normal at the time of the initial instrument oscillation. There were no facilities damaged or radioactive releases associated with the sensed level indications and the automatic safety features functioned as required.

Cause or Causes - The root cause of the three items of concern described above is attributed to serious deficiencies in management controls of licensed activities.

For the first item, a series of major deficiencies in management controls resulted in a protracted failure of the Pilgrim facility to comply with the provisions of 10 CFR 50.44. When 10 CFR 50.44 became effective, the containment atmosphere control system actually installed at Pilgrim Station did not meet all of the regulatory requirements. This condition existed due to management's failure to conduct a proper design review of the capabilities of the existing atmosphere control system. However, the licensee erroneously informed the NRC in a letter dated October 19, 1979 that the existing installed equipment in Pilgrim Unit 1 was in full compliance with the requirements of 10 CFR 50.44. However, apparently as a result of an October 30, 1979 NRC letter requesting details of Pilgrim's compliance with 10 CFR 50.44, the licensee took steps to design and install a modification to the system which would bring Pilgrim into compliance. This modification was installed during the May 1980 outage; however, because of a failure of management to initiate an essential procedural change, the modified system was not fully operational until June 5, 1981.

Prior to installation of the system modification, the failure of licensee management to properly determine system capabilities via a thorough design analysis of the installed system (as compared with the requirements of 10 CFR 50.44) led to the erroneous report to the NRC in October 1979. Further, when the licensee subsequently discovered in early 1980 that the installed system did not meet the requirements of 10 CFR 50.44, the licensee did not so inform the NRC and correct the material false statement made in the October 19, 1979 letter.

For the second item, the case involved a breakdown in the control of planned maintenance activities. There was a failure to properly review and control safety-related activities at the facility. The reduction in the level of safety was discovered and identified to the licensee's staff by the NRC Resident Inspector at the site.

For the third item, the problem of apparent erroneous level oscillations was determined to be caused by flashing of the level instrument reference legs at reduced reactor pressure because of the high drywell operating temperature (240°F) which was in excess of that specified in the FSAR (150°F). Drywell temperatures higher than this specified limit are attributed to ineffective drywell cooling due to a degraded condition of the drywell ventilation system (ducting, coolers, cooling water). The high drywell temperatures and degraded condition of the cooling systems had been observed by Pilgrim station operating personnel on many previous occasions and are considered to have been allowed to continue as a result of inadequate preventive maintenance and management controls in this area.

Actions Taken to Prevent Recurrence

Licensee - For the first item, the licensee restored the system to its original design and initiated an investigation to determine the cause of the unauthorized maintenance. Also, a procedural revision was made to permit effective remote operation of the system. The licensee proposed, and the NRC approved, technical specification changes concerning operability and surveillance requirements of the modified hydrogen control system.

For the second item, when the NRC Resident Inspector discovered the deficiency and notified the licensee, the licensee restored the partially disabled containment isolation control logic electrical circuits to a fully operable condition.

For the third item, corrective maintenance was initiated on the drywell cooling systems to restore the original design capacity during the refueling outage which began on September 26, 1981. Drywell equipment insulation was repaired and additional instrumentation was installed to monitor the drywell temperature and performance of the cooling systems. At the request of the NRC, the licensee proposed Technical Specifications limiting drywell temperatures. In addition, the licensee conducted special inspections, tests, and evaluations for possible detrimental effects on safety-related equipment subjected to this sustained abnormally high temperature environment. Certain equipment, such as instrument limit switches, electrical cables, and solenoids were found to be affected and were either repaired or replaced. The licensee submitted safety evaluations and actions to the NRC for review.

In response to the NRC Order, Notice of Violation and Proposed Imposition of Civil Penalties (Ref. 15), described below under NRC actions, the licensee paid the civil penalty in full on March 19, 1982 (Ref. 16). In response to the Order, on March 18, 1982 the licensee submitted to the NRC for review and approval a Performance Improvement Program (Ref. 17) which involves a comprehensive action plan of tasks and milestones to correct the deficiencies identified in the NRC Order. The program includes an independent appraisal of site and corporate organizations and functions, modifications in organizational structure, improvements to be made in management control and oversight systems, and programs designed to improve individual performance. The proposed program would span a time period of from 18 to 24 months; however, both the scope and schedule of the program may be affected by the findings of the independent appraisal.

The licensee selected a contractor for the independent appraisal and the appraisal was initiated on February 23, 1982. The contractor's work is to be reviewed by a peer review group including executives from other utilities operating BWRs.

The licensee has restructured corporate functions within its Nuclear Organization. All nuclear activities at Boston Edison have been reassigned to an organizational unit directed by a Senior Vice President who has no other function or line responsibilities. Also, a new position of Director-Nuclear Operations Review having responsibility for corporate management oversight of on-site safety related activities was established and filled.

The licensee has several other improvement tasks in progress, including: (1) a corporate on-site review program, (2) evaluation of safety review and assessment functions, (3) monitoring of commitments, (4) monitoring of changes to NRC regulations, (5) a corrective action system, (6) completeness and accuracy of reporting to the NRC, (7) 10 CFR 50.59 review requirements, (8) quality assurance and preventative maintenance programs, and (9) training programs.

NRC - Based on the first two items, and previous deficiencies in regulatory performance, the NRC concluded that continued operation of the plant over the long term required significant changes in the control of licensed activities. As a result, the NRC issued an Order Modifying Licensee Effective Immediately on January 18, 1982 (Ref. 15) requiring Boston Edison Company to develop and submit for NRC review and approval a comprehensive plan of action that will yield an independent appraisal of site and corporate management controls and oversight, and a review of previous safety-related activities to evaluate compliance with NRC requirements. Concurrent with the Order, the NRC issued a Notice of Violation and Proposed Imposition of Civil Penalties in the amount of $550,000.

As discussed previously, the licensee submitted a Performance Improvement Program in response to the NRC Order. In a letter to the licensee on April 23, 1982 (Ref. 18), the NRC Region I staff found the licensee's Performance Improvement Program acceptable subject to progress reviews at predetermined milestones and periodic meetings with Boston Edison Company management.

The NRC (Region I) conducted a special investigation between November 24, 1981 and January 7, 1982 to evaluate the circumstances surrounding the material false statement submitted to the NRC in the licensee's October 19, 1979 letter. This investigation also examined the licensee's failure to notify the NRC when it was subsequently discovered by the licensee's staff that the requirements of 10 CFR 50.44 had not been fully met. The investigation concluded that the material false statement was not deliberate and that contrary information subsequently developed by the licensee's staff was not intentionally withheld from the NRC; both of these resulted from a lack of effective management control of licensee communication with the NRC. The results of this investigation were transmitted to the licensee on March 18, 1982 (Ref. 19).

For the third item, the NRC conducted inspections and reviewed the licensee's corrective actions. An NRC meeting was held on December 18, 1981 where licensee representatives described their plans and schedules for resolution of this major problem. At this meeting, the licensee was directed to propose Technical Specifications limiting drywell temperatures and provide a safety evaluation which describes the basis for operations with drywell temperatures exceeding maximum design values. This item was included in a citation for violations in a letter from NRC Region I to the licensee dated February 4, 1982 (Ref. 14).

The NRC has approved safety evaluation reports submitted by the licensee for the modified containment atmosphere control system and for past operation at elevated drywell temperatures. The NRC has agreed that the modified containment atmosphere control system and maintenance actions to replace components possibly degraded by the high drywell temperature meet regulatory requirements. The NRC has also approved technical specifications submitted by the licensee which limit drywell temperature during plant operation. The Pilgrim facility recovered from the protracted refueling (September 1981 - March 1982) and achieved criticality on March 26, 1982.

NRC Region I has expanded the inspection program at Pilgrim to more thoroughly evaluate continuing licensee performance in light of the problems identified with management control. Through the inspection program and periodic management meetings, NRC Region I will closely follow implementation of the licensee's Performance Improvement Program.

This incident is closed for purposes of this report.

* * * * * * *

82-4 Steam Generator Tube Rupture at R. E. Ginna Nuclear Power Plant

Preliminary information pertaining to this event was reported in the Federal Register (Ref. 20). Appendix A (Example 2 of "For Commercial Nuclear Power Plants") of this report notes that major degradation of the primary coolant pressure boundary can be considered an abnormal occurrence.

Date and Place - At 9:28 a.m. on January 25, 1982, the R. E. Ginna Nuclear Power Plant experienced a reactor trip as a result of a steam generator tube rupture.

At 9:33 a.m., the operating staff at the plant notified the NRC Headquarters Operations Center of the event (via the Emergency Notification System phone). The R. E. Ginna plant utilizes a Westinghouse designed pressurized water reactor (PWR). The plant is owned and operated by Rochester Gas and Electric Corporation (the licensee) and is located in Wayne County, New York.

Nature and Probable Consequences - Steam generator tubes in a pressurized water reactor are an integral part of the reactor coolant pressure boundary. The inner part of the tubes contain the reactor coolant fluid while the outer parts (or shell side) contain the feedwater which, when heated, becomes steam which is then piped to drive the turbine generators. Thus, the loss of integrity of steam generator tubes results in a breach of the primary-to-secondary system boundary.

Safety-margins are maintained through conservative design, inservice inspections, and administrative controls during operation such that if a steam generator tube leaks, the leakage can be detected rapidly and the reactor can be shut down safely. Periodic inspections are required to assure that defective steam generator tubes are detected and removed from service. When a tube is found to be defective or leaking, integrity of the steam generator is restored by either plugging the tube at both ends or repaired by a sleeving process. The repair reduces the likelihood of rapid leaks developing or of defective tubes existing which could result in adverse safety consequences if they fail alone during normal operations or as a consequence of certain postulated accidents. In spite of these design and administrative controls, the rupture of a steam generator tube can happen and did occur at Ginna, and previously at Point Beach Unit 1, Surry Unit 2 and Prairie Island Unit 1. This is one of the design basis accidents considered in the NRC safety review of nuclear power plants.

Pressurized water reactor nuclear power plant licensees are required to have operational plans (including procedures, trained operation and support personnel, and other capabilities) to cope with a complete rupture of a steam generator tube and mitigate any radiological consequences. The R. E. Ginna operating and support staff mitigated the consequences of the January 25 event such that the radiological consequences were insignificant in terms of risk from any resultant on-site or off-site exposures.

Facility Response

A summary of the facility response during the incident is described in the Annex at the end of this abnormal occurrence. This section, and the following sections pertaining to radiological assessment, institutional response, and post-event activities, were generally extracted from NUREG-0909, the NRC Task Force report of the event (Ref. 21). Figure 3 shows a schematic diagram of the plant systems in normal operation.

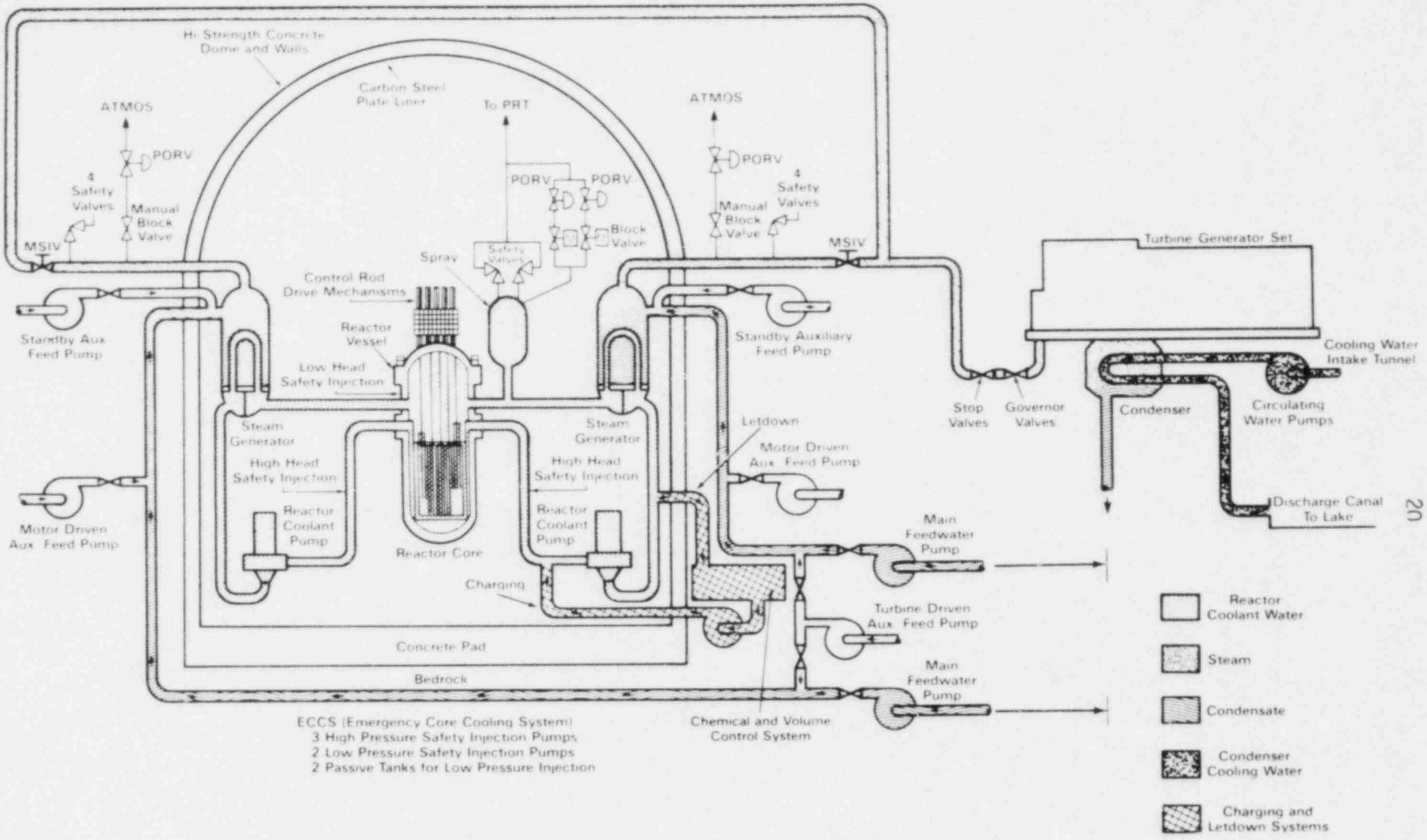

Figure 3 Schematic diagram of the Ginna Nuclear Power Plant systems in normal operation.

Radiological Assessment

The radiological consequences were assessed by the NRC from the estimated curies available for release from the reactor coolant system, the amount of activity transferred to the faulted steam generator, and the activity released to the environment as a function of time. Both airborne and liquid releases were estimated. Airborne release figures were then converted to projected off-site dose figures using conservative dispersion models based on existing weather conditions. On- and off-site radionuclide release and exposure measuring devices were also read and the results analyzed. The risk to the public and licensee personnel was then estimated.

In summary, most radionuclides released from Ginna were released during the first three hours of the event. During this period the wind was blowing toward the southeast. Snow and moist cold air caused a large fraction of the radio-iodines and particulates released from Ginna to be deposited on the Ginna site rather than to remain airborne beyond the site boundary. Offsite releases during the event were estimated to be less than 25% of the limit for unrestricted areas. It is estimated that airborne releases to a licensee-controlled, unrestricted area (onsite, adjacent to Ginna Brookwood Training Center) exceeded 10 CFR 20 limits for average yearly airborne concentrations. However, all releases resulted in doses far less than 10 CFR 100 guidelines, which are established as the design basis for accident conditions.

With regard to occupational radiation exposures on the day of the event, some plant personnel incurred radiation doses in the course of performing routine duties and/or while responding to the event. The maximum recorded individual occupational radiation dose on the day of the event was 240 millirems as compared with the limits specified in 10 CFR 20 of 1250 millirems for a three-month period for radiation workers (or 3000 millirems under certain specified conditions). The health risk to the maximally exposed invididuals onsite and offsite from exposure to radioactive materials released at Ginna is considered to be much less than the risk from exposure to any of the major sources of radiation (e.g., medical exposure and natural background radiation) and within the same range as the risks from exposure to many of the other common sources of enhanced radiation exposure (e.g., from airline travel, natural gas heat, and television viewing).

It was estimated that a total of about 90 curies of noble gases were released, mostly from the steam jet-air ejector. About 0.4 curies of dose-equivalent I-131 (a total of about 5 curies of all isotopes of iodine), and about 1.3 curies of cobalt, molybdenum, barium, and cesium are estimated to have been released, mostly from the openings of the safety valve. About 25 curies of tritium may have been released, mostly from the safety valve openings with trace amounts from the air ejector.

The estimated figure from noble gases released is believed to be conservative due to the large volume of highly radioactive gases discovered in the faulted steam generator after the event. Unfortunately, these gases were removed from the generator before their curie content was accurately determined.

Institutional Response

Various organizations including the licensee, State and local governments, NRC, and other Federal agencies responded to the event at Ginna.

The licensee had primary responsibility for resolving the conditions that existed at the plant. Prescribed initial notifications to the NRC and to authorities of the State and local counties were completed very early in the event, and interaction throughout the event occurred among all the participants.

The Nuclear Regulatory Commission, using the resources of the Senior Resident Inspector, the Region I Base and Site Teams, and the Headquarters Executive and Analytical Teams, monitored the licensee's actions in response to the event to assure that these actions were correct and appropriate.

The State of New York and Wayne and Monroe Counties were promptly notified by the licensee. They responded by activating their Emergency Operations Centers and by sending representatives to the site. Monroe County also fielded off-site radiological monitoring teams and reported results back to the Emergency Operations Center throughout the day. Twice during the first day of the event, the Governor of New York was briefed by the Chairman of the NRC on the status of the event.

The Federal Emergency Management Agency (FEMA) was notified of the event by NRC. FEMA then coordinated the Federal agency nontechnical response, both from their Headquarters and from their Region II facility in New York City. Agencies contacted by FEMA were: the U.S. Department of Agriculture, the Department of Energy, the Coast Guard, the Department of Housing and Urban Development, the Department of Transportation, the National Oceanic and Atmospheric Administration, the Environmental Protection Agency, the General Services Administration, the Department of Health and Human Services, and the Department of Defense. Each of the agencies notified was prepared to respond in accordance with the responsibilities defined in the National Radiological Emergency Preparedness/Response Plan. FEMA also kept the White House advised of developments. News media interest was very high and the event received extensive coverage.

Post-Event Activities

After the plant was placed in cold shutdown, the licensee established conditions to support the identification of the ruptured tube in the affected steam generator. On January 31, the plant staff completed purging hydrogen and other noncondensible gases which had accumulated in the B steam generator as a result of the reactor coolant system inleakage to the waste gas system. On February 1, the plant staff completed a reactor coolant system and B steam generator drain-down procedure. No major problems were encountered.

The licensee determined by a hydrostatic test that the ruptured tube was located at row 42, column 55 (R42C55) on the hot-leg of the steam generator. Nondestructive examinations of this tube, including eddy-current, radial profilometry, fiber optics, and visual inspections showed that the rupture was about 4 inches long and 0.7 inches wide at its center. The rupture was centered about 5 inches above the tubesheet. The rupture was fish-mouth-shaped and pointed outward along tube column 55. Figure 4 shows an artist's sketch of two closesup views of the tube rupture.

TV-optics examination inside the B steam generator identified damage to additional tubes that had been plugged previously because of eddy-current indications, leakage, or their proximity to other plugged tubes. In addition, a number of foreign objects were found and removed from the secondary side of the faulted steam generator. The most significant object found was approximately pie-shaped, roughly 4.18 inches wide by 6.31 inches long by 0.5 inches thick; the object had the same appearance and metallic characteristics as part of the steam generator downcomer flow resistance plate. The latter plate had been cut into pieces and reportedly had been removed during a steam generator modification in 1975. In addition, some previously plugged tubes displayed evidence of gross mechanical damage, and at least two of these tubes were fractured and found skewed between the tube bundle and the steam generator shell. Some small foreign objects were also found in the A steam generator.

A visual examination of the ruptured tube showed evidence of classical fretting wear with transverse scoring. There was also evidence of previous oxidized wear markings. The wall thickness at the rupture point was about 5% of the original thickness and the tube appeared ballooned at the rupture location.

Cause or Causes - Based on extensive inspections, test, and analyses, the licensee has postulated that a large foreign object in the steam generator initiated a sequence of events which eventually led to the tube rupture. Ineffective quality control practices during steam generator modifications in 1975 and subsequent modifications resulted in foreign objects falling (and remaining undetected) into the tubesheet in the downcomer region outside the periphery of the tube bundle. The postulated failure mechanism is that foreign objects, in conjunction with normal thermal and hydraulic loadings during steam generator operation, impacted on the outermost peripheral tubes causing damage. During a later inspection, these tubes were plugged based on eddy-current indications and/or small leaks. However, foreign objects continued to damage the plugged tubes until eventually some collapsed and in some cases severed. These severed, plugged tubes damaged adjacent tubes, whether plugged or unplugged. These adjacent, damaged, unplugged tubes were, in turn, plugged as a result of eddy-current indications or leaks. However, the damage mechanism continued to occur until some of these tubes also became severed. Eventually, tube R42C55 was damaged by an adjacent tube which had been plugged previously and which subsequently severed. The wear on tube R42C55 occurred relatively uniformly over several inches of length such that local penetration of the wall and small leakage did not occur before the tube became sufficiently weakened to rupture.

Actions Taken to Prevent Recurrence

Licensee - The licensee performed extensive evaluations of the tube rupture event. A report summarizing the sequence of events, operator actions, emergency procedures, equipment performance, radiological assessment, and recommendations for future actions was submitted to the NRC by letter dated April 13, 1982. A

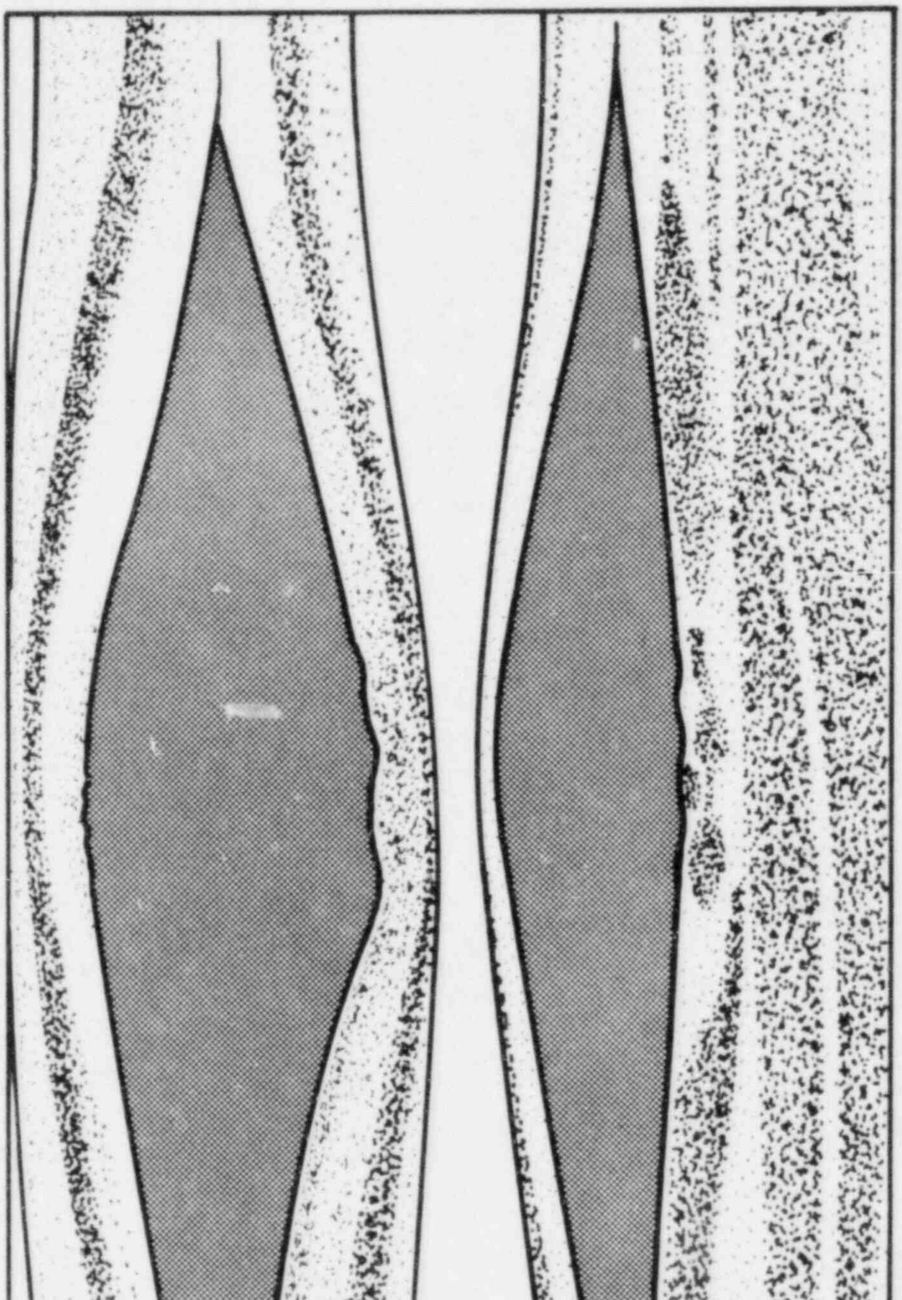

Figure 4 Two closeup views of rupture in tube R42C55

second report, covering the steam generator inspection, evaluation, and repair program was submitted to the NRC by letter dated April 26, 1982 (Ref. 22).

The licensee identified a number of actions to be taken, both prior and after restart of the plant, to upgrade equipment and procedure deficiencies identified in the evaluaton of the event. Equipment upgrading included changes to the wide-range pressure instrumentation for the reactor coolant system, letdown system isolation, reactor coolant loop subcooling meter, and pressurizer PORV air control system. Procedure changes included clarifying action statements and their bases, clarifying when certain actions can be taken, and adding guidance to resolve other deficiencies identified in responding to the tube rupture event.

The licensee's steam generator evaluation program objectives were to determine the full extent of defects, determine the tube failure mechanism(s), restore the steam generator to a condition which is safe to operate, and to obtain NRC concurrence for return to power. The extent of defects was determined by extensive eddy current inspection of tubes and fiber optics, video and visual inspections of the secondary side. The failure analysis program consisted of metallurgical examinations, analyses, and testing of damaged tube samples removed from the B steam generator. The postulated failure mechanism, based on this program, is discussed under "Cause or Causes" above. The repair program included plugging the cold leg of the ruptured tube, removing metallurgical samples of selected tubes, removing structurally degraded tube sections, removing foreign objects from both steam generators, and performing a series of inspections and tests after repairs to assure that the steam generator would be ready for return to service. After plant restart, an intermediate outage would be scheduled later to perform another set of steam generator inspections to assure that the corrective actions taken to preclude further peripheral tube defects have been successful. In addition, the licensee has installed a loose parts monitoring system on both steam generators which will help assure that any loose objects are detected in the future.

NRC - The NRC responded to the event by activating its Incident Response Centers in Headquarters and Region I, sending a Site Team to the plant site, and coordinating with the licensee on technical support matters and with other organizations on emergency preparedness and public information matters. After the event, a Task Force was formed to gather information, assess it, and prepare a report of factual information associated with the event. The report was issued in April 1982 and is designated NUREG-0909 (Ref. 21). The report provides the required data base for additional detailed analysis within the NRC which could lead to further actions. The report also contains a listing and discussion of the significant findings from the investigation of the event and associated response actions.

The NRC reviewed the licensee's evaluation of the event and the proposed corrective actions. After resolution was obtained (Ref. 23), the NRC agreed with the licensee that the plant could be restarted and taken to full power. The reactor achieved criticality on May 25, 1982. The NRC safety evaluation report related to the restart of the plant was issued as NUREG-0916 (Ref. 24).

This incident is closed for purposes of this report.

Annex: Facility Response

The sequence of events for the steam generator tube rupture incident and the associated response actions during the incident are described below. Also described are assessments of the amount of fluid released from the faulted steam generator and possible adverse effects on the pressure vessel due to the temperature transient of the reactor coolant system. This Annex was generally extracted from NUREG-0909 (Ref. 21).

Prior to the tube rupture, the plant was operating at 100% power with normal operating temperature and pressure. Figure 3 shows a schematic diagram of the plant systems in normal operation. No indications of primary-to-secondary leakage existed. On January 25, 1982, at 9:25 a.m., multiple control room alarms alerted the operators to a reactor coolant system (RCS) rapid depressurization. The air ejector radiation monitor alarm indicated to them the existence of a steam generator tube rupture; other alarms indicated the rupture was probably in the B steam generator. The operators commenced manual actions including a rapid turbine power reduction and an increase in the number and speed of the operating charging pumps. At 9:28 a.m., the continuing reactor coolant system pressure drop resulted in an automatic reactor trip and an automatic safety injection actuation causing all three high pressure safety injection pumps to start. As a result of safety injection actuation, an automatic containment isolation occurred and the operating charging pumps automatically tripped. All safety systems operated as required. Both reactor coolant pumps were manually stopped and the operators then verified that natural circulation cooling had developed in both reactor coolant system loops. The pressurizer emptied and the reactor coolant system initial depressurization reached a minimum of about 1200 psig. Briefly, during the initial depressurization transient, a small steam bubble formed in the upper head during natural circulation. This bubble subsequently collapsed as safety injection flow refilled the reactor coolant system.

Initially, operators cooled down the plant by sending steam from both steam generators to the main condenser, while they confirmed the identify of the faulted steam generator. The B steam generator was isolated at about 9:40 a.m., and natural circulation in the B loop terminated shortly thereafter. Although all sources of feedwater to the B steam generator had been isolated, its water level continued to rise because of the flow through the tube rupture (break flow). At 9:55 a.m., the narrow-range water level indicator on the B steam generator went off-scale high and subsequently the B main steam line started to fill.

At 9:57 a.m., the safety injection actuation circuitry was reset to allow the resetting of the containment isolation system. After containment isolation was reset, instrument air to the containment and, therefore, control of the air-operated valves inside containment, was restored.

At 10:07 a.m., operators attempted to equalize the pressure differential between the reactor coolant system and the B steam generator to stop the flow through the tube rupture by opening a pressurizer power operated relief valve (PORV). This PORV was operated successfully three times. During its fourth operation, the valve opened on command but when the operator placed its controls in the closed position, the valve started to close then reopened and remained in the open position. When the operator noticed that the PORV had failed to close, he manually closed its block valve to prevent the further depressurization of the reactor coolant system and further loss of coolant out the open PORV. During these operations, the pressurizer level had risen rapidly and the level instrument was now indicating off-scale high.

Operation of the pressurizer PORV resulted in the formation of steam bubbles in the reactor vessel upper head region and in the top of the tubes in the B steam generator. The size of the bubble in the reactor vessel upper head region was estimated to be about 300 ft^3. The total bubble volume in the steam generator tubes was smaller. The growth of these steam bubbles during the depressurization of the reactor coolant system, along with increased safety injection flow, had caused the rapid filling of the pressurizer. Natural circulation in the A loop and core cooling were not adversely affected by the existence of these bubbles.

A B steam generator code safety valve lifted and closed three times as a result of continued break flow into the B steam generator; however, the safety valve may have leaked steam starting after the first lift. At 10:38 a.m., safety injection was terminated to prevent further safety valve lifts.

At 10:40 a.m., the condensate system was shut down to prevent further radioactive contamination of the condensate storage tanks and the condensate demineralizers. The original contamination had resulted by the dumping of steam to the condenser from the faulted B steam generator earlier in the vent. To continue the plant cooldown, the operators vented the A steam generator to atmosphere using its PORV.

At about 10:52 a.m., the rupture disc on the pressurizer relief tank (PRT) burst as a result of inventory additions from three sources. A letdown line relief was the major contributor, with the pressurizer PORV and reactor coolant pump seal return line relief also adding water to the pressurizer relief tank.

At 11:07 a.m., one safety injection pump was started to provide a buffer for the anticipated drop in reactor coolant system pressure that the plant staff expected to occur as a result of the restart of the A reactor coolant pump. Again, at 11:19 a.m., a B steam generator safety valve lifted and closed; however, by this time the steam line had flooded sufficiently to cause water rather than steam to be released. At about 11:21 a.m., the A reactor coolant pump was started. The resulting coolant flow cooled and collapsed any remaining steam bubbles in the reactor vessel upper head region and the B steam generator. At about 11:37 a.m., a fifth lift of the B steam generator safety valve occurred and the safety injection pump was stopped. The safety valve closed, but apparently continued to leak water at about 100 gpm.

At 11:52 a.m., the indicated pressurizer level returned onscale as a result of the continued break flow from the reactor coolant system into the B steam generator. Because pressurizer heaters were reenergized before the restart of the A reactor coolant pump, a steam bubble had been reestablished in the pressurizer. At 12:02 p.m., normal letdown from the reactor coolant system to the chemical and volume control system was reestablished. Because the B steam generator safety valve continued to leak, the tube rupture continued to drain the reactor coolant system to the B steam generator. The rate of decrease of pressurizer level resulting from the continued flow thorugh the break prompted the operators to restart one safety injection pump. This was done at about 12:12 p.m. This pump was intermittently run to control pressurizer level until about 12:35 p.m. The B steam generator safety valve apparently stopped leaking at about 12:28 p.m., so the safety injection pump was not needed to control pressurizer level after this point.

At 12:27 p.m., the reactor coolant system and B steam generator pressures equalized. The operators then maintained indicated reactor coolant system pressure about 25 psi below B steam generator pressure to promote backflow through the tube break. At 6:40 p.m., the B steam generator water level returned on-scale on the narrow-range indicator. The B steam generator was then cooled by a feed-and-bleed operation with auxiliary feedwater being intermittently supplied to the B steam generator while backflow through the break was allowed to continue.

At 7:00 a.m., January 26, the residual heat removal system was placed in operation. At 6:53 p.m., the same day, the licensee declared the plant to be in a cold shutdown condition.

The rate of flow through the ruptured tube was estimated by the NRC using mathematical models and known plant system characteristics. By these estimates, the highest flow rate through the break occurred very near the onset of the rupture and was calculated to be about 760 gpm. A mass balance algorithm indicated that about 117,000 pounds of steam and water were released from the B steam generator.

The indicated temperature transient of the reactor coolant system loops was reviewed by the NRC in an attempt to determine whether the reactor vessel had experienced a significant thermal shock during the event. Using the available plant parameter data and the information known about the design of the plant, the thermal behavior of the reactor coolant system while the safety injection pumps were operating and the reactor coolant pumps were stopped was modeled. The results of calculations based on the model paralleled closely the actual transient experienced. These calculations, and later analysis (Ref. 24) completed after NUREG-0909 (Ref. 21) was issued, indicate that a significant thermal shock did not occur during this event.

FUEL CYCLE FACILITIES

(Other than Nuclear Power Plants)

The NRC is reviewing events reported by these licensees during the first calendar quarter of 1982. As of the date of this report, the NRC has not determined that any 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 first calendar quarter of 1982. As of the date of this report, the NRC has not determined that any were abnormal occurrences.

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

REFERENCES

1. U.S. Nuclear Regulatory Commission, "Abnormal Occurrence: Diesel Generator Engine Cooling System Failures," Federal Register Vol. 47, No. 142, July 23, 1982, 31998-32002.

2. Confirmaton of Action Letter from James G. Keppler, Regional Administrator, NRC Region III, to Cordell Reed, Vice President, Commonwealth Edison Company, Docket Nos. 50-237 and 50-249, November 20, 1981.*

3. Confirmation of Action Letter from James G. Keppler, Regional Administrator, NRC Region III, to Cordell Reed, Vice President, Commonwealth Edison Company, Docket Nos. 50-10, 50-237, and 50-249, November 25, 1981.*

4. Conformation of Action Letter from James G. Keppler, Regional Administrator, NRC Region III, to Cordell Reed, Vice President, Commonwealth Edison Company, Docket Nos. 50-237 and 50-249, December 2, 1981.*

5. Letter from C. E. Norelius, Director, Divison of Emergency and Technical Inspection, NRC Region III, to Cordell Reed, Vice President, Commonwealth Edison Company, forwarding Inspection Report No. 50-237/81-35 and No. 50-249/81-27, Docket Nos. 50-237 and 50-249, January 22, 1982.*

6. U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 82-08, "Check Valve Failures on Diesel Generator Engine Cooling System," March 26, 1982.*

7. U.S. Nuclear Regulatory Commission, "Abnormal Occurrence: Pressure Transients During Shutdown at a Nuclear Power Plant," Federal Register (Item is being published in Federal Register concurrently with this report).

8. U.S. Nuclear Regulatory Commission, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from Director, NRR to NRR Staff"; USNRC Report NUREG-0138, published November 1976.***

---

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

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

***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.

9. U.S. Nuclear Regulatory Commission, "Report to Congress on Abnormal Occurrences: July - September 1976," USNRC Report NUREG-0090-5, published March 1977.***

10. Letter from J. P. O'Reilly, Regional Administrator, NRC Region II, to R. E. Uhrig, Vice President, Advanced Systems and Technology, Florida Power and Light Company, "Report Nos. 50-250/81-32 and 50-251/81-31," February 2, 1982.*

11. U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 82-17, "Overpressurization of Reactor Coolant System," June 11, 1982.*

12. U.S Nuclear Regulatory Commission, "Abnormal Occurrence: Major Deficiencies in Management Controls at a Nuclear Power Plant," Federal Register Vol. 47, No. 115, June 15, 1982, 25793-25796.

13. Letter from R. C Haynes, Regional Administrator, NRC Region I, to J.E. Howard, Vice President, Nuclear, Boston Edison Company M/C Nuclear, "Inspections 50-293/81-8 and 50-293/81-22," Docket No. 50-293, January 19, 1982.*

14. Letter from R. W. Starostecki, Director, Division of Resident and Project Inspection, NRC Region I, to William D. Harrington, Senior Vice President, Nuclear, Boston Edison Company M/C Nuclear, "Inspections 50-293/81-24 and 50-293/81-35," Docket No. 50-293, February 4, 1982.*

15. Letter from R. C. DeYoung, Director, NRC Office of Inspection and Enforcement, to F. M. Staszesky, President, Boston Edison Company M/C Nuclear, fowarding a "Notice of Violation and Proposed Imposition of Civil Penalties" and an "Order Modifying License Effective Immediately," Docket No. 50-293, January 18, 1982.*

16. Letter from F. M. Staszesky, President, Boston Edison Company, to R. C. DeYoung, Director, NRC Office of Inspection and Enforcement, Docket No. 50-293, March 19, 1982.*

17. Letter from W. D. Harrington, Senior Vice President, Nuclear, Boston Edison Company, to R. C. Haynes, Regional Administrator, NRC Region I, Docket No. 50-293, March 18, 1982.*

18. Letter from R. C. Haynes, Regional Administrator, NRC Region I, to William D. Harrington, Senior Vice President, Nuclear, Boston Edison Company M/C Nuclear, Docket No. 50-293, April 23, 1982.*

19. Letter from R. C. Haynes, Regional Administrator, NRC Region I, to F.M. Staszesky, President, Boston Edison Company M/C Nuclear, "NRC Investigation Report No. 50-293/81-37," Docket No. 50-293, March 18, 1982.*

20. U.S. Nuclear Regulatory Commission, "Abnormal Occurrence: Steam Generator Tube Rupture at R.E. Ginna Nuclear Power Plant," Federal Register Vol. 47, No. 135, July 14, 1982, 30672 - 30676.

21. U.S. Nuclear Regulatory Commission, "NRC Report on the January 25, 1982 Steam Generator Tube Rupture at R.E. Ginna Nuclear Power Plant," USNRC Report NUREG-0909, April 1982.**

22. Letters from John E. Maier, Vice President, Electric and Steam Production, Rochester Gas and Electric Corporation, to D. M. Crutchfield, Chief, Operating Reactors Branch No. 5, NRC Nuclear Reactor Regulation, Docket No. 50-249, April 13, 1982 and April 26, 1982. Supplemented by letters dated April 23, April 26, May 5, May 6, May 17, May 18, and May 21, 1982.*

23. Letters from D. M. Crutchfield, Chief, Operating Reactors Branch No. 5, NRC Nuclear Reactor Regulation, to John E. Maier, Vice President, Electric and Steam Production, Rochester Gas and Electric Corporation, "Restart of R. E. Ginna Nuclear Power Plant," Docket No. 50-244, May 22, 1982 and May 25, 1982.*

24. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report related to restart of R. E. Ginna Nuclear Power Plant," USNRC Report NUREG-0916, May 1982.**

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:

<u>For All Licensees</u>

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 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.

<u>For Commercial Nuclear Power Plants</u>

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).

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 Part 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 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.

APPENDIX B

UPDATE OF PREVIOUSLY REPORTED ABNORMAL OCCURRENCES

During the January through March 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

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; and Vol. 4, No. 4. It is further updated as follows:

79-3 Nuclear Accident at Three Mile Island

Reactor Building Entries

The licensee completed 30 reactor building (RB) entries during the first quarter of 1982. Most of those were in support of the reactor building decontamination experiment which was conducted during the month of March.

Five RB entries took place in January. The major task during these entries was the installation of support for an electrically powered lift which would be used to transport personnel and equipment from the refueling floor to the polar crane. Additional task included the testing of source range neutron monitor, NI-2, sampling for iodine 129, area radiation surveys, and videotaping of areas in the RB.

Nine RB entries were made during the month of February and the following tasks were accomplished:

--a load test of the polar crane mounted supports for a power lift

--assembly of the power lift platform on the 347 ft. elevation of the RB

--installation of new fire hoses on RB fire stations

--power lift installation/load test

--portable, gamma spectrometer survey of the 347 ft. elevation

--installation of decontamination hoses through penetration R-561

In February, the installation of systems necessary to support the gross decontamination experiment was also completed.

During March, 16 entries were made in the RB to support the gross decontamination experiment which was conducted on various levels and surfaces within the TMI-2 reactor building. The primary purpose of the experiment was to evaluate techniques for reducing radiation levels due to surface contamination by flushing with water and other means. Accident water that has been processed through the Submerged Demineralizer/EPICOR II systems was used for the flushing. The Polar Crane, D-Rings, Missile Shields, Refueling Canal, Refueling Bridge, equipment and surfaces of the 305' and 345' 6" elevations were flushed with low pressure water. In addition, the floor surfaces on elevation 305' and the floor surfaces and major pieces of equipment on elevation 347' 6" were sprayed with high pressure water. Water pressure at the tip of the spray nozzle ranged from 1000 psig for low pressure flushes to about 6000 psig for high pressure flushes with temperatures that ranged from ambient to 140°F. Another portion of the experiment included the application of strippable coatings and chemical decontamination solutions on limited surfaces in the reactor building. These tasks included:

--Decontamination of a 150 ft^2 section of the reactor building floor using a detergent and mechanical scrubber,

--Decontamination of a 150 ft^2 section of the reactor building floor using a phosphoric acid solution and a mechanical scrubber,

--Decontamination of a 500 ft^2 section of the reactor building floor using a strippable coating.

The decontamination experiment was concluded with the following activities:

--Portable gamma spectrometer survey of the 305 ft. and 347 ft. elevations,

--Concrete bore samples collection,

--Radiation survey of polar crane, and

--Still and motion picture photography.

Work in the reactor building consisted of 280 man-hours of preparatory and related work (e.g., data acquisition, polar crane inspection, health physics, maintenance, spider shafter placement) which incurred a cumulative occupational dose of about 66 man-rem. Actual water flushing required about 140 man-hours and resulted in a cumulative occupational dose of about 30 man-rem, while approximately 10,000 gallons of processed accident water was added to the sump.

The effectiveness of the gross decontamination experiment is currently being evaluated.

Submerged Demineralizer System (SDS)

Processing of reactor building sump water continued through the first quarter of 1982 with only minor operational problems. Processing of batch 17 which commenced on January 20, 1982, was immediately secured when a process train leak into the spent fuel pool water was detected. (The SDS process train is located underwater in the Unit 2 spent fuel pool.) Plant monitors showed no increase in effluents to the environment as a result of the leak. Subsequently, it was determined that the use of the leak was a worn gasket which seals the connection between the system piping and the ion exchange vessels. The gasket was replaced and after a satisfactory leak check processing was resumed on January 23.

The radioactivity in the spent fuel pool water increased from approximately 1×10^{-4} µCi/ml (gross β-γ activity) to 7×10^{-3} µCi/ml. The increase did not result in increased levels of airborne radioactivity and, thus, did not affect the health and safety of workers in the vicinity of the spent fuel pool. Further, the increase did not significantly interrupt further processing of water through the SDS.

After the processing of Batch 23 was completed, during the first week in March, the SDS was secured for minor maintenance. To date, approximately 600,000 gallons of sump water have been processed by the SDS and polished by EPICOR-II. The reactor building sump has about 30,000 gallons of water remaining at a depth of approximately 5 inches. (Originally, water in the sump was 8½ feet deep.) The licensee is developing plans for the removal of this remaining water since the floating pump now in the sump loses suction at approximately 5 inches.

The licensee is also directing engineering efforts towards preparation for processing the Reactor Coolant System (RCS) through the SDS. None of the 90,000 gallons of water in the RCS have been processed to date. The basic operation of the SDS will remain the same as during the processing of the reactor building sump water except that the effluent will be directed to a reactor coolant bleed tank in preparation for injection back into the RCS instead of the EPICOR II for polishing. Since the RCS cannot be completely drained, a feed and bleed batch operation to and from the reactor coolant bleed tanks will be used to remove water from the RCS in preparation for processing. This feed and bleed type operation will result in effectively processing approximately 300,000 gallons through the system because the same water recirculates several times. Since the effluent of the SDS is directed back into the RCS, actual additions to the total inventory of processed water at TMI are minimized and the volume of RCS water will remain at about 90,000 gallons. The chemistry of the water returned to the RCS will be closely controlled to reduce the possibility of corrosion of materials in the RCS.

RCS processing is currently scheduled to commence during the month of June. However, this schedule is uncertain due to current financial constraints.

Advisory Panel

The NRC's Advisory Panel for the Decontamination of TMI Unit 2 met on January 28, 1982, in Harrisburg. Status reports were given by GPU, NRC, EPA, DOE, and comments were offered by several members of the public. EPA representatives stated that their Agency was not reducing operations at TMI despite nationwide reductions in many of EPA's other programs. Dauphin County Commissioner Larry Hochendoner supported decoupling the Unit 1 startup from the Unit 2 cleanup by applying the Deferred Energy Surcharge which is due to expire in June 1982, toward the cleanup. Issues pertaining to financial aspects of the Cleanup were also a topic of discussion.

Subsequently, on March 23, 1982, the TMI Advisory Panel met with the NRC Commissioners in Washington, DC. The Advisory Panel asked the Commission to increase its efforts to resolve the political/financial problems that are delaying the cleanup. The Commission has written to U.S. Senator James McClure, Chairman of the Energy Subcommittee, Senator Alan Simpson, Chairman of the Subcommittee on Nuclear Regulation, and Representative Morris Udall, Chairman of the Subcommittee on Energy and the Environment, advancing its position that "greater federal participation in assuring financial viability is a prerequisite to an acceptably rapid (cleanup) program."

Advisory Panel Chairman John Minnich, in transmitting the Panel's second report to Chairman Palladino on March 17, 1982, concluded that there is as yet no significant financial commitment to fund the cleanup. Mr. Minnich added his personal recommendation that the Commission hold a meeting on TMI issues in the Harrisburg area to hear public comments. The Panel's official report also concluded that "given the long-term serious hazards posed by TMI-2," the Commission should do all it can to ensure the cleanup proceeds expeditiously.

The citizen's advisory panel made several suggestion to the Commissioners at the March 23 meeting that are being considered, including:

1. Urge the White House to assign a high-level liaison to deal with TMI cleanup problems;

2. Attempt to clarify the utility companies' (EEI) proposal to contribute $192 million to TMI cleanup funding.

Miscellaneous Items

Particulate Radioactivity Increase in Unit 2 Auxiliary and Fuel Handling Buildings

At 9:53 a.m. on January 8, 1982, the licensee declared an Unusual Event as a result of an indication of increased airborne radioactivity in the Unit 2 Auxiliary and Fuel Handling Buildings. Personnel working in the buildings were immediately evacuated. The licensee's final investigation of this event indicated that the increased airborne radioactivity in the building resulted from blowdown of the service air lines into a potentially contaminated floor drain.

The licensee reported a slightly increased indication on the plant airborne effluent monitor (HPR-219). Initial off-site dose calculations indicated a minute fraction of a millirem, at the site boundaries, which is indistinguishable from natural background. An off-site survey team took measurements at about 0.5 mile south of the TMI Visitor's Center and confirmed no detectable radioactivity offsite. The possible source of this airborne activity was isolated at 10:00 a.m. All in-plant monitors and effluent monitors were trending down to normal readings at 10:48 a.m. The licensee terminated the Unusual Event at 11:30 a.m.

Tritium Increase in Ground Water Samples

Tritium levels in water samples from several test borings in the vicinity of the borated water storage tank (BWST) increased substantially in February 1982. The licensee subsequently increased the sampling frequency of all test borings.

The tritium levels detected were below the maximum permissible concentration for unrestricted areas and posed no hazard to workers or the general public. The increase in ground water activity resulted from a leak of BWST water on January 13, 1982, caused by a frozen, cracked valve. The estimated 50 gallons of leakage contained the following isotopic concentrations:

H^3	1.1×10^{-1} μCi/ml
Sr^{90}	2.9×10^{-6} μCi/ml
Cs^{134}	1.6×10^{-4} μCi/ml
Cs^{137}	3.6×10^{-4} μCi/ml

Since soil absorbs and slows the migration of the non-tritium nuclides to varying degrees, the potential for detection of these nuclides in future well samples exists.

Tritium concentrations in subsequent ground water samples remained above their January 1982 levels through March. However, water samples taken after February 11, 1982 indicated that concentrations of all other isotopes except tritium had declined below the lower limit of detection.

Apparent Oxygen Deficient Atmosphere in the Reactor Building

On February 19, 1982 an Unusual Event was declared at TMI Unit 2. This action was precipitated at the start of the routine reactor building entry when selected instruments indicated the following potential problems:

1) Low O_2 reading (17%) in the RB,

2) Offscale high readings on some of the portable radiation monitoring equipment,

3) Indication of the presence of combustible gases in the RB and,

4) Indication of excessive hydrogen content (1 - 1.5% H_2) in the RB.

A common mode failure of instruments was suspected but the continuous inability to confirm this led to the precautionary judgment that the Unusual Event should be declared.

Following the analysis of RB air sample results, the problem with the high combustible and low oxygen readings was subsquently attributed to instrument malfunctions. Radiofrequency interference, battery undercharging/failure and sensor cell deterioration all contributed to the false alarms and readings.

Purification System Filter Removal

During the week of March 21, a team of eight technicians removed the remaining four filters from the letdown and makeup systems (purification system). (Two of the filters had been removed from the system in 1981.) Specifically, filters upstream and downstream of the purification demineralizers and filters on the discharge of the high pressure makeup pumps were removed. Following the filter removals, technicians used vacuum cleaners to remove debris from the filter housings. These filters are currently scheduled to be shippped in April to a DOE contractor for analysis.

However, Monday, March 22, 1982, the licensee declared an Unsual Event for nearly 2 hours when control room operators noticed that excessive makeup was going to the reactor coolant system during the filter removal operation. It was determined that the problem was caused by leakage through a reach rod operated filter isolation valve. When technicians opened the filter housing, approximately 200 gallons of makeup system (Standby Pressure Control system) water spilled onto the floor of the filter cubicle. Technicians subsequently closed other valves in the system which isolated the valve that leaked, and the leak rate began to decrease.

The water that leaked out of the system was vacuumed off the floor into 55 gallon drums and drained to the auxiliary building sump for processing. The highest exposure dose rate measured on contact with any of the drums was 350 mr/hr. Operators later rerouted the 90 psig makeup water flow and closed additional valves in series to stop the leak. The total cumulative personnel dose for the filter removal and cleanup operation was approximately one man-rem.

* * * * * * *

The following abnormal occurrence was originally reported in NUREG-0090, Vol. 4, No. 3, "Report to Congress on Abnormal Occurrences: July-September 1981." It is upated as follows:

81-4 Failure of High Pressure Safety Injection System

The modified Safety Injection System was tested in accordance with the accelerated surveillance schedule applicable during this operating cycle on November 24, 1981 and February 27, 1982. In each case the valves operated satisfactorily, with no measurable increase in opening force and no apparent "set in" effect.

The licensee has developed a schedule for procurement and installation of eight replacement valves. According to this schedule, installation would be completed in the first half of 1985. One significant factor in the extended schedule is the time required for inspection and testing of the valves (14 months after completion of fabrication). The licensee is also performing an engineering study on an alternative system which would utilize dedicated safety injection pumps, and thereby eliminate some of the complexities inherent in the present design. If the licensee determines that such a redesign is warranted, the necessity to replace the present valves will be considered as part of the redesign effort.

Further reports will be made as appropriate.

* * * * * * *

The following abnormal occurrence was originally reported in NUREG-0090, Vol. 4, No. 4, "Report to Congress on Abnormal Occurrences: October-December 1981." It is updated as follows:

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

The independent design reverification program for Diablo Canyon is being performed in two phases. The first phase (Phase I) involves the reverification of seismic design activities performed prior to June 1978. Phase II will involve 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. Teledyne has submitted a comprehensive plan to the NRC which details how Phase I of the reverification would be performed. This plan is currently under review.

The reverification program to date has identified approximately 140 open items, including six items which have been classified as "errors." The significance of the "errors" is being assessed by the NRC staff.

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.

1. Low Concentrations of Tritium Detected in Groundwater at Sheffield Low-Level Waste Disposal Facility

On February 9, 1982, during routine sampling of ground water, the U.S. Geological Survey (USGS) detected concentrations of tritium in two of the 17 wells located east of the Sheffield (Illinois) Low-Level Waste Disposal Facility. The wells were constructed under contract to the NRC and are on private property approximately 200 feet east of the site. The concentrations were approximately 90 nanocuries (nCi) per liter in one and 60 nCi/liter in the other. Maximum permissible concentration established in 10 CFR Part 20 for liquid releases of tritium is 3,000 nCi/liter. There is no drinking water taken from the aquifer where the tritium was detected. Although the concentrations, because of their magnitude, pose no threat to public health and safety, the Illinois Department of Nuclear Safety and the NRC determined that the cause and areal extent of the migration should be investigated.

State and Federal officials were notified by the USGS of the presence of tritium and confirmatory samples were taken from all 17 wells. These samples confirmed the presence of tritium in the two wells in question. No tritium was found in the other off-site wells east of the site.

Migration of tritium in small concentrations is an expected occurrence in the vicinity of low-level waste disposal areas. From a technical standpoint the migration is of interest because it affords the opportunity to learn more about the mechanisms of radionuclide migration and to better predict the level and extent of future migration in order to determine what, if any, mitigating measures are warranted.

The State of Illinois, through the Attorney General's Office, petitioned the site licensee, US Ecology, Inc., to construct additional wells and perform studies necessary to determine the cause, extent, and magnitude of off-site tritium concentrations in ground water. On February 16, 1982, the 13th Circuit Court of Illnois issued an Agreed Order which stated that US Ecology, Inc., would fund such a study up to $100,000. Subsequently, an ad hoc task force was established to oversee the study, consisting of NRC, Illinois Department of Nuclear Safety, Illinois State Geological Survey, USGS, Illinois State Attorney General's Office, and US Ecology, Inc., personnel. The additional studies are being performed in two phases, the first of which was completed in June, 1982. Phase I consisted of the installation of eight ground water observation wells in the vicinity of previously observed migration. Results of the testing of these wells were discussed at a task force meeting on June 10, 1982. Measurable amounts of tritium were found in two of these additional wells, one on site, and one on private property east of the site. Concentrations in these wells were approximately 30 and 4 nCi/liter, respectively. Thus, a total of four wells in a narrow band near the northeast corner of the site have been found to have measureable tritium concentrations. In addition, geophysical surveys to determine the extent of sand deposits through which the tritium is migrating were performed. Based on these results, Phase II planning has been completed. Phase II will consist of approximately eight additional wells to confirm the likely source and lateral extent of the migration. Drilling of these wells is scheduled to begin in August. Testing of the wells should be completed in September. Final results of the study are expected by early fall 1982.

There has been local interest in the tritium migration. Newspaper accounts and opinions have appeared in both local and county papers. At the request of Bureau County officials, a public information meeting was held in Princeton, Illinois, on February 18, 1982. State and Federal officials attended the meeting to explain the situation and to discuss actions to be taken. A second public meeting was held in Princeton by Illinois officials on June 7, 1982 to discuss the history of the Sheffield site.

Since the tritium concentrations detected were only a small fraction of the maximum permissible concentration established in 10 CFR Part 20, there was no appreciable impact on public health and safety. Therefore, this event is not considered to be reportable as an abnormal occurrence.

NRC FORM 335 (7-77)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER (Assigned by DDC)
NUREG-0090, Vol. 5, No. 1

2. (Leave blank)

3. RECIPIENT'S ACCESSION NO.

4. TITLE AND SUBTITLE (Add Volume No., if appropriate)
Report to Congress on Abnormal Occurrences
January - March 1982

5. DATE REPORT COMPLETED
MONTH | YEAR

7. AUTHOR(S)

DATE REPORT ISSUED
MONTH August | YEAR 1982

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
U.S. Nuclear Regulatory Commission
Office for Analysis and Evaluation of Operational Data
Washington, D.C. 20555

6. (Leave blank)

8. (Leave blank)

12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
U.S. Nuclear Regulatory Commission
Office for Analysis and Evaluation of Operational Data
Washington, D.C. 20555

10. PROJECT/TASK/WORK UNIT NO.

11. CONTRACT NO.

13. TYPE OF REPORT
Quarterly

PERIOD COVERED (Inclusive dates)
January - March 1982

15. SUPPLEMENTARY NOTES

14. (Leave blank)

16. ABSTRACT (200 words or less)

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 January 1 to March 31, 1982.

During the report period, there were four abnormal occurrences at the nuclear power plants licensed to operate. The first involved diesel generator engine cooling system failures. The second involved pressure transients during shutdown. The third involved major deficiencies in management controls. The fourth involved a steam generator tube rupture. There were no abnormal occurrences for the other NRC licensees during the report period. The Agreement States reported no abnormal occurrences to the NRC.

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

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT
Unlimited

19. SECURITY CLASS (This report)
Unclassified

21. NO. OF PAGES

20. SECURITY CLASS (This page)
Unclassified

22. PRICE
$

NRC FORM 335 (7-77)

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

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

FOURTH CLASS MAIL
POSTAGE & FEES PAID
USNRC
WASH. D. C.
PERMIT No. G67

120555078877 1 ANCOCYNQ
US NRC
ADM DIV OF TIDC
POLICY & PUBLICATIONS MGT BR
PDR NUREG COPY
LA 212
WASHINGTON DC 20555

NUREG-0090, Vol. 5, No. 1

REPORT TO CONGRESS ON ABNORMAL OCCURRENCES
JANUARY - MARCH 1982

AUGUST 1982