NUREG-0090
Vol. 5, No. 2

Report to Congress on Abnormal Occurrences

April - June 1982

U.S. Nuclear Regulatory Commission

Office for Analysis and Evaluation of Operational Data

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ABSTRACT

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event which the Nuclear Regulatory Commission determines to be significant from the standpoint of public health or safety and requires a quarterly report of such events to be made to Congress. This report covers the period from April 1 to June 30, 1982.

The report states that for this report period there were no abnormal occurrences at the nuclear power plants licensed to operate. There were no abnormal occurrences for the other NRC licensees. The Agreement States reported no abnormal occurrences to the NRC.

The report also contains information updating some previously reported abnormal occurrences. Some of the updates have been given more generalized titles (as compared to their former more specific titles) to include some new events which are associated in some respects to previously reported abnormal occurrences. As of the date of the report, these events have not been determined reportable as abnormal occurrences. The items which have been retitled are discussed in Appendix B of the report.

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PREFACE

INTRODUCTION

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

Events are currently identified as abnormal occurrences for this report by the NRC using the criteria delineated in Appendix A. These criteria were promulgated in an NRC policy statement which was published in the Federal Register on February 24, 1977 (Vol. 42, No. 37, pages 10950-10952). In order to provide wide dissemination of information to the public, a Federal Register notice is issued on each abnormal occurrence with copies distributed to the NRC Public Document Room and all local public document rooms. At a minimum, each such notice contains the date and place of the occurrence and describes its nature and probable consequences.

The NRC has reviewed Licensee Event Reports, licensing and enforcement actions (e.g., notices of violations, civil penalties, license modifications, etc.), generic issues, significant inventory differences involving special nuclear material, and other categories of information available to the NRC. The NRC has determined that only those events, including those submitted by the Agreement States, described in this report meet the criteria for abnormal occurrence reporting. This report covers the period between April 1 to June 30, 1982.

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

THE REGULATORY SYSTEM

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

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

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

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

REPORTABLE OCCURRENCES

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

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

Information concerning reportable occurrences at facilities licensed or otherwise regulated by the NRC is routinely disseminated by NRC to the nuclear industry, the public, and other interested groups as these events occur. Dissemination includes deposit of incident reports in the NRC's public document rooms, special notifications to licensees and other affected or interested groups, and public announcements. In addition, a computer printout containing information on reportable events received from NRC licensees is routinely sent to the NRC's more than 100 local public document rooms throughout the United States and to the NRC Public Document Room in Washington, D.C.

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

AGREEMENT STATES

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

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

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

FOREIGN INFORMATION

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

REPORT TO CONGRESS ON ABNORMAL OCCURRENCES

APRIL-JUNE 1982

NUCLEAR POWER PLANTS

The NRC is reviewing events reported at the nuclear power plants licensed to operate during the second calendar quarter of 1982. As of the date of this report, the NRC had not determined that any events were abnormal occurrences.

FUEL CYCLE FACILITIES

(Other than Nuclear Power Plants)

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

OTHER NRC LICENSEES

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

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

The NRC is reviewing events reported by these licensees during the second calendar quarter of 1982. As of the date of this report, the NRC had not determined that any events 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 second calendar quarter of 1982, the Agreement States reported no abnormal occurrences to the NRC.

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 April through June 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. Some of the updates below have been given more generalized titles (as compared to their former more specific titles) to include some new events which are associated in some respects to previously reported abnormal occurrences. As of the date of the report, these events have not been determined reportable as abnormal occurrences. The reasons for retitling are discussed below in the introductory material for the specific items affected.

NUCLEAR POWER PLANTS

The following abnormal occurrence was originally reported in NUREG-75/090, "Report to the Congress on Abnormal Occurrences: January-June 1975," and updated (and previously closed out) in subsequent reports in this series, i.e., NUREG-0090-1; 0090-2; 0090-3; Vol. 1, No. 3; Vol. 2, No. 2; Vol. 2, No. 4; Vol. 3, No. 2; and Vol. 3, No. 4. It is being reopened to report the following case of pipe cracking which will require a significant plant outage to repair.

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

On March 23, 1982 the Niagara Mohawk Power Corporation (the licensee) reported an event involving leakage from welds on two nozzles connecting recirculation system piping to the reactor vessel of Nine Mile Point Unit 1. The plant, which utilizes a General Electric-designed boiling water reactor, is located in Oswego County, New York.

The leakage was discovered during performance of a routine hydrostatic pressure test prior to return to operation from a scheduled maintenance outage. At a test pressure of 900 psig, leakage was observed near two of the ten recirculation-piping-to-reactor vessel nozzles. Upon depressurization and removal of thermal insulation, three small leaks were observed on the Loop #11 discharge nozzle safe end and one leak on the Loop #15 suction nozzle safe end. Each of the leaks appeared to be in the heat-affected zone of the nozzle safe-end-to-pipe weld. Samples were obtained from one of the safe ends in the vicinity of the throughwall cracks and sent to General Electric and Battelle Laboratories for evaluation. The results confirmed the presence of intergranular stress corrosion cracking.

Inspections then were made of the recirculation pump discharge casting to riser elbow welds. Again cracks were found. Testing by General Electric and

Sylvester Associates confirmed the presence of intergranular stress corrosion cracking. The licensee decided to inspect all of the remaining welds where radiation fields permitted. The results of those examinations indicated cracking in a large number of welds.

Based on the licensee's examinations and investigations, the licensee decided to replace the 28-inch recirculation piping in all five recirculation loops and all ten safe ends (Ref. B-1). If warranted, the branch piping would also be replaced. The replacement material will be of a type less susceptible to intergranular stress corrosion cracking.

On September 21, 1982, The NRC issued Inspection and Enforcement Information Notice No. 82-39 to all boiling water reactor licensees to inform them of this event (Ref. B-2).

The replacement program will require extensive work. The plant is expected to be shut down for at least a year. Even with the reactor core removed and the reactor coolant system drained and decontaminated, it is preliminarily estimated that a collective dose to workers of over 2000 person-rems will be accumulated. More precise numbers will be developed as details of the replacement program are better defined.

The NRC will closely monitor the licensee's corrective actions--performing inspections, meeting with the licensee, requesting additional information from the licensee, and making evaluations--to assure that the plant will be safe to restart. In addition, conditions of the license require various reports to allow NRC monitoring of worker dose as work proceeds. The generic implications of the event are being considered and a Bulletin may be issued.

This incident is closed for purposes of this report.

*   *   *   *   *   *   *

76-2 <u>Occupational Overexposure During Entry to Reactor Cavity Area</u>

This item was previously reported and closed out in NUREG-0090-3 ("Report to Congress on Abnormal Occurrences, January-March 1976") with the title of "8 Rem Occupational Whole Body Exposure." It is being reopened and retitled to report an incident similar to the March 19, 1976 incident reported in NUREG-0090-3, that occurred at the same plant (Zion Unit 1). The plant, which is operated by Commonwealth Edison Company, utilizes a pressurized water reactor and is located in Lake County, Illinois.

As described below, the March 25, 1982 incident resulted in an occupational overexposure. The Commission is concerned due to the potential for very high exposures to radiation (over 1000 R/hr in some areas) in the reactor cavity area when irradiated incore detector thimbles are retracted in the area. Though the licensee's controls were insufficient to prevent an occupational overexposure, the overexposure received was less than the abnormal occurrence reporting threshold.

On March 25, Zion Unit 1 was in a scheduled refueling outage and attempts were being made to flood the refueling pool. However, several leaks were experienced principally around cover gaskets in the refueling pool floor which provide access to the excore nuclear instrumentation. The way the licensee verified leakage was by entering the reactor cavity area, the area below the reactor vessel and refueling pool. This area is normally locked since incore detectors are withdrawn and stored and associated incore thimbles are withdrawn.

The health physicist on duty authorized an exposure of 500 mrem for a shift engineer to make the inspection for leakage. Two rad/chem technicians (one was a trainee) were assigned by their foreman to assist the shift engineer (one to survey and one to keep time). The details of the proposed inspection were not discussed.

After obtaining the required radiation protection clothing and full face masks, the three people entered the radiation zone above the reactor cavity. This area contained the entrance opening to a ladder extending down to the reactor cavity. One rad/chem technician climbed down the ladder first to survey the dose rate at the bottom. It was about 50 R/hr. The rad/chem technician did not leave the ladder to survey since there were about six inches of water on the floor of the cavity, and he assumed the shift engineer would not leave the ladder. The second rad/chem technician (the trainee) stayed at the top of the ladder to keep time. He was told that timekeeping would begin when he was called to start by the first rad/chem technician. The timekeeper then was supposed to call down to the first technician when necessary to keep the shift engineer within his 500-mrem approved dose, and the first rad/chem technician would notify the shift engineer to come back up the ladder.

After timekeeping began, however, the shift engineer left the base of the ladder and moved forward on the grating into an area which had not been surveyed for radiation levels. He walked approximately six to eight feet away from the ladder. The timekeeper was unaware that the shift engineer had left the base of the ladder, and calculated the exposure using 50 R/hr. After approximately 30 seconds, the timekeeper signaled the other rad/chem technician to call the shift engineer back. The rad/chem technician did so, waited approximately 10 seconds, and called again. The timekeeper stopped timekeeping when he saw the shift engineer on the ladder. The total timekeeping period was 67 seconds. The shift engineer's total exposure was calculated by the timekeeper to be 931 mrem (50 R/hr x 67 sec).

The health physicist who had authorized the exposure was informed that the shift engineer may have been in an area where the exposure rates exceeded 50 R/hr. The film badge was sent to a contractor for emergency processing on March 26. The next day, the contractor called in a reading of 3700 mrem. This reading was double-checked and confirmed. This exposure is in excess of the limits specified in 10 CFR 20.101b.

Further evaluation indicated that the dose received was about 5 rem, considering that the highest dose to the whole body was at the knee and that the film badge was worn between the chest and the waist.

As reported in the licensee's written report of the event (Ref. B-3), the licensee's corrective actions included: (1) requiring: that thimbles be reinserted, prior to personnel entry into the reactor cavity beyond the base of ladders extending into the cavity area; that a special lock be placed on the door to the cavity when the thimbles are removed; and that locations of thimbles and incore detectors during outages be posted in the rad/chem office; (2) reviewing and supplementing operations and radiation personnel training; (3) requiring a radiation work permit (which includes a written description of work to be performed) for individual jobs exceeding 50 mrem; (4) revising radiation protection procedures to include specific requirements for issuing high-range (500 mrem, and over) dosimeters; and (5) maintaining in the rad/chem department additional high-range detectors calibrated to 1000 R/hr, and a limited number of dose rate ionization chamber instruments with lighted dials for work in dark areas.

The results of an NRC inspection conducted on March 30-31, April 7-8, and April 29, 1982 indicated serious weaknesses in the licensee's radiation protection program for systematic evaluation and planning of radiation work. Specific weaknesses identified in the NRC enforcement letter (Ref. B-4) included: (1) lack of coordination between plant health physicists and rad/chem foremen in planning entry into high radiation level areas, (2) inadequate radiation surveys associated with the entries, (3) use of inexperienced rad/chem technicians to monitor the entries, (4) lack of understanding by radiation protection personnel of the reactor cavity radiological hazards including the radiation sources, (5) inadequate training in reactor cavity radiological hazards even though a similar overexposure had occurred in 1976, (6) failure of shift operations personnel in leadership positions to exhibit good radiation protection practices, and (7) unavailability of survey instruments calibrated to greater than 50 R/hr.

On July 9, 1982, the NRC issued a Notice of Violation and Proposed Imposition of Civil Penalties of $100,000 to the licensee because of what the NRC characterized as a serious breakdown in management controls and the radiation protection program, resulting in an unnecessary exposure (Ref. B-4). The NRC was particularly concerned that this overexposure occurred even though the licensee had implemented actions to avoid such an occurrence based on the similar overexposure at Zion Unit 1 in 1976 and in response to the NRC's IE Circular No. 76-03, "Radiation Exposures in Reactor Cavities," issued on September 13, 1976 (Ref. B-5). It is noted that a similar event also occurred at Salem Unit 1 which was summarized in the NRC publication <u>Power Reactor Events</u>, Vol. 3, No. 2 (Ref. B-6).

The licensee replied to the NRC enforcement letter on August 9, 1982 (Ref. B-7). The licensee paid the civil penalty and admitted to the items of noncompliance. The licensee also responded to the weaknesses identified by the NRC in the enforcement letter. The NRC is reviewing the licensee's specific responses and may take additional action if necessary.

This incident is closed for purposes of this report.

* * * * * * *

The following abnormal occurrence was originally reported in NUREG-0090-5, "Report to Congress on Abnormal Occurrences: July-September 1976," and updated in subsequent reports in the series, i.e., NUREG-0090-8; Vol. 1, No. 4; Vol. 2, No. 3, Vol. 2, No. 4; Vol. 3, No. 1; Vol. 3, No. 2; Vol. 3, No. 4; and Vol. 4, No. 1. It is further updated as follows:

76-11 Steam Generator Problems

As reported in NUREG-0090, Vol. 3, No. 4, reporting on the progress of the generic studies on steam generator tube integrity is provided annually in the section of the NRC's Annual Report (Ref. B-8) which addresses Unresolved Safety Issues, and quarterly in NUREG-0606, "Unresolved Safety Issues Summary, Aqua Book" (Ref. B-9). Reports on unique oprerating experience or problems with the various aspects on steam generators, as a component, will be made as appropriate in these quarterly abnormal occurrence reports to Congress (NUREG-0900 series) under this section, which has been retitled from "Steam Generator Tube Integrity" to "Steam Generator Problems."

Experience as of November 1981

A general update of steam generator tube experience as of November 1982 was issued by the NRC in NUREG-0886 (Ref. B-10). The following summary is based on the report.

PWR steam generators have been experiencing a variety of tube degradation problems for a number of years, caused by either corrosion or mechanical damage. Degradation experience prior to the mid-1970's included wastage (localized thinning of tube walls) and caustic stress corrosion cracking (SCC) on the secondary side due to difficulties in adequately controlling the chemistry of secondary water with sodium phosphate and to impurities carried into the steam generators by the feedwater. The establishment of all volatile treatment (AVT) control succeeded in arresting any further significant wastage, but SCC has continued as a concern, particularly in plants with a significant period of phosphate operation prior to conversion to AVT. All PWRs have been converted to or have operated exclusively with AVT control except Robinson Unit 2 and San Onofre Unit 1.

Another form of degradation is denting, which is the squeezing of tubes caused by the buildup of corrosion products between the tubes and their carbon steel supports. This results in tube leaks due to stress corrosion cracks initiating from the primary side. At least 23 Westinghouse and Combustion Engineering PWRs have reported denting, including 8 where denting is considered extensive. Copper oxide has been demonstrated to be a catalyst, and plants with copper in their secondary cycles have increased susceptibility to denting. The earliest date for a plant using all ferritic stainless steel supports, which will reduce the susceptibity to denting, is 1983. The B&W design of antivibration supports with minimum contact area, along with virtually copper-free systems and full flow condensate polishing, have combined to maintain once-through steam generators free of significant denting to date.

Cracking has occurred in the small radius, inner row (row 1) U-bends in Westinghouse steam generators. Some of this is associated with denting, which leads to support plate deformation and eventual closure of the support plate flow slots. At Surry Unit 2, this caused a tube rupture in 1976. The current industry practice of plugging all row-1 tubes on the observance of upper flow slot closure has prevented similar failures in other extensively dented steam generators. In some cases, cracks have been observed where there has been no denting, and this is attributed to residual stresses introduced during the process of fabricating the tubes.

Corrosion of steam generator tubes in the crevices between the tube and the tubesheets has been experienced in at least 7 of the 17 Westinghouse plants where the tubes were not expanded to full depth of the 24-inch thick tubesheet. General intergranular attack has also been reported at two units in a region of sludge accumulation on the tubesheet. A new pitting phenomemon has recently been observed at Indian Point Unit 3 and at Millstone Unit 2, affecting large numbers of tubes. Localized wall thinning at the tube supports on the cold leg has been observed since 1979 at Prairie Island Unit 2, affecting over 100 tubes. (After NUREG-0886 was issued, a similar problem was found at Salem Unit 1 in January 1982.) Robinson Unit 2, which continues to operate with phosphate secondary water chemistry control, has also experienced local wall thinning in the U-bends, which possibly is phosphate wastage related.

Leakage in the new Westinghouse Model D steam generators at Ringhals Unit 3 in Sweden caused its shutdown in October 1981. Significant tube wall reductions in the preheater section of that plant and of Almaraz Unit 1 in Spain have been indicated by eddy current testing. This was caused by wearing down of tube walls from vibrational rubbing against baffle plates. To date, McGuire Unit 1 is the only domestic operating plant with Model D steam generators, and the licensee pursued a cautious power escalation and test program with frequent shutdowns for inspection to determine susceptibility to this vibration phenomenon (this problem is discussed further below).

Until recently, the principal degradation modes of B&W units involved circumferential fatigue cracking and erosion-corrosion. However, extensive corrosion-induced cracking on the primary side in the upper tubesheet region has recently been observed at Three Mile Island Unit 1 (this problem is discussed further below).

Steam generator problems are generally detected by inservice inspections and primary-to-secondary leakage of coolant. Corrective actions generally involve plugging of the degraded tubes or the use of a sleeving process. The advantage of sleeving is that it permits the tube to remain in service. Extensive tube degradation can lead to significant downtime to perform steam generator inspections. Eventually, a sufficient number of tubes may be plugged such that the plant must be derated. To avoid these problems, some utilities have elected either to replace their severely degraded steam generators or are considering doing so; e.g., Virginia Electric Power Company's Surry Units 1 and 2, Florida Power and Light Company's Turkey Point Units 3 and 4, and Carolina Power and Light Company's Robinson Unit 2.

As concluded in NUREG-0886, steam generators manufactured by each of the three PWR vendors have experienced various forms of tube degradation resulting from a combination of inadequate design and fabrication, nonoptimal secondary system design and construction material and poor operating practices, especially in secondary water chemistry control and condenser maintenance. In addition, the inspection, repair, and replacement efforts needed to deal with these problems have also resulted in radiation exposures which account for a major portion of each facility's annual occupational radiation dose. Industry-sponsored research has helped to identify the causes and mechanisms for several different types of tube degradation phenomena which has subsequently led to some design and operating improvements. It is anticipated that tube degradation will continue, but at a slower rate primarily because of better controls of variables leading to the problems rather than because of corrections to design deficiencies and construction materials. Although some steam generator vendors have recently developed new steam generator models that are expected to provide significantly greater margins against tube degradation during operation, all plants scheduled to received an operating license before 1984 have steam generators similar to those currently in service.

The PWR vendors, the affected utilities, and the NRC staff are continuing to evaluate new areas where the potential for tube degradation exists and to improve condenser integrity, secondary water chemistry control, steam generator and secondary plant designs and non-destructive inspection capabilities to minimize forced outages caused by steam generator tube failures. The NRC staff has been evaluating adverse experience on a case-by-case basis and has concluded that continued operation and licensing do not constitute an undue risk to the health and safety of the public.

Experience from November 1981 through August 1982

Some of the more important problems encountered since issuance of NUREG-0886 are described below.

Tube Rupture

On January 25, 1982, the R. E. Ginna Nuclear Power Plant experienced a steam generator (Westinghouse designed) tube rupture while the plant was operating at 100% power. The single tube rupture was the worst experienced to date resulting in a calculated flow rate through the rupture of about 760 gpm. Licensees are required to have operational plans to cope with such an event, and as such, the licensee (Rochester Gas and Electric Corporation) mitigated the consequences of the January 25, 1982 event so that the radiological consequences were insignificant in terms of risk from any resultant on-site or off-site exposures. Details of the event have been previously reported in an NRC Task Force (formed as a result of the event) report (Ref. B-11). In addition, since the event was considered a major degradation of the primary coolant pressure boundary, it was reported as abnormal occurrence AO 82-4 in NUREG-0090, Vol. 5, No. 1.

Steam Generator Auxiliary Feedwater Header Damage in Certain B&W-Designed Plants

A unique problem, resulting in steam generator tube leakage was discovered at Davis-Besse Unit 1 on April 13 1982. On that day, Toledo Edison Company notified the NRC resident inspector that evidence of auxiliary feedwater (AFW) header damage was observed in the No. 1 steam generator of Davis-Besse Unit 1. Later inspections also showed damage to the header of the No. 2 steam generator. Subsequent inspections on April 19-20, 1982 and April 29, 1982 at Rancho Seco Unit 1 (operated by Sacramento Municipal Utility District) and Oconee Unit 3 (operated by Duke Power Company), respectively, showed similar damage to the AFW headers. Davis-Besse Unit 1, Rancho Seco Unit 1, and Oconee Unit 3 all utilize Babcock & Wilcox (B&W)-designed pressurized water reactors and are located in Ottawa County, Ohio; Sacramento County, California; and Oconee County, South Carolina, respectively.

Description of AFW Header Design

The auxiliary feedwater system is used to provide emergency heat removal capability upon loss of the normal feedwater to the secondary side (outside surface of the tubes) of the steam generators. B&W-designed plants utilize "once-through" steam generators (S/G), one in each reactor coolant loop. Reactor coolant water flows from the reactor, through the loop hot leg to the tube (primary) side of the S/G where heat is transferred to the shell (secondary) side to produce steam which drives the turbine generator to produce electricity.

Auxiliary feedwater (AFW) enters a distribution header, which for B&W-designed S/Gs is located either inside or outside the S/G. External AFW headers are used at the following B&W-designed plants which have accumulated operational experience of over 22 reactor years: Oconee Units 1 and 3, Arkansas Nuclear One Unit 1, Crystal River Unit 3, and Three Mile Island Unit 1 (shut down since the accident at Three Mile Island Unit 2 on March 28, 1979). B&W utilized an internal header design at Davis-Besse Unit 1, Rancho Seco Unit 1, Oconee Unit 3, Three Mile Island Unit 2 (indefinitely shut down since the March 28, 1979 accident), and Midland Units 1 and 2 (both units are still under construction).

The internal AFW header design is described as follows (Ref. B-12); Figure 1 shows a longitudinal section of the design for Davis-Besse Unit 1 (Rancho Seco Unit 1 and Oconee Unit 3 designs are similar). The internal AFW header is a rectangularly shaped torus fabricated of welded plate segments. The header is positioned on the upper end of the upper vertical cylindrical baffle (upper shroud). The header also serves as a continuation of the upper shroud to separate the tube bundle from the steam annulus. The header is positioned and retained by eight sets of inner and outer brackets welded to the bottom of the header and match drilled through the shroud. A dowel pin passes through each set of brackets and is welded to the inner bracket (see Fig. 1).

A single 3-1/2-inch-diameter AFW nozzle delivers water to the header via a thermal sleeve which fits into the header. Water leaves the header through 60

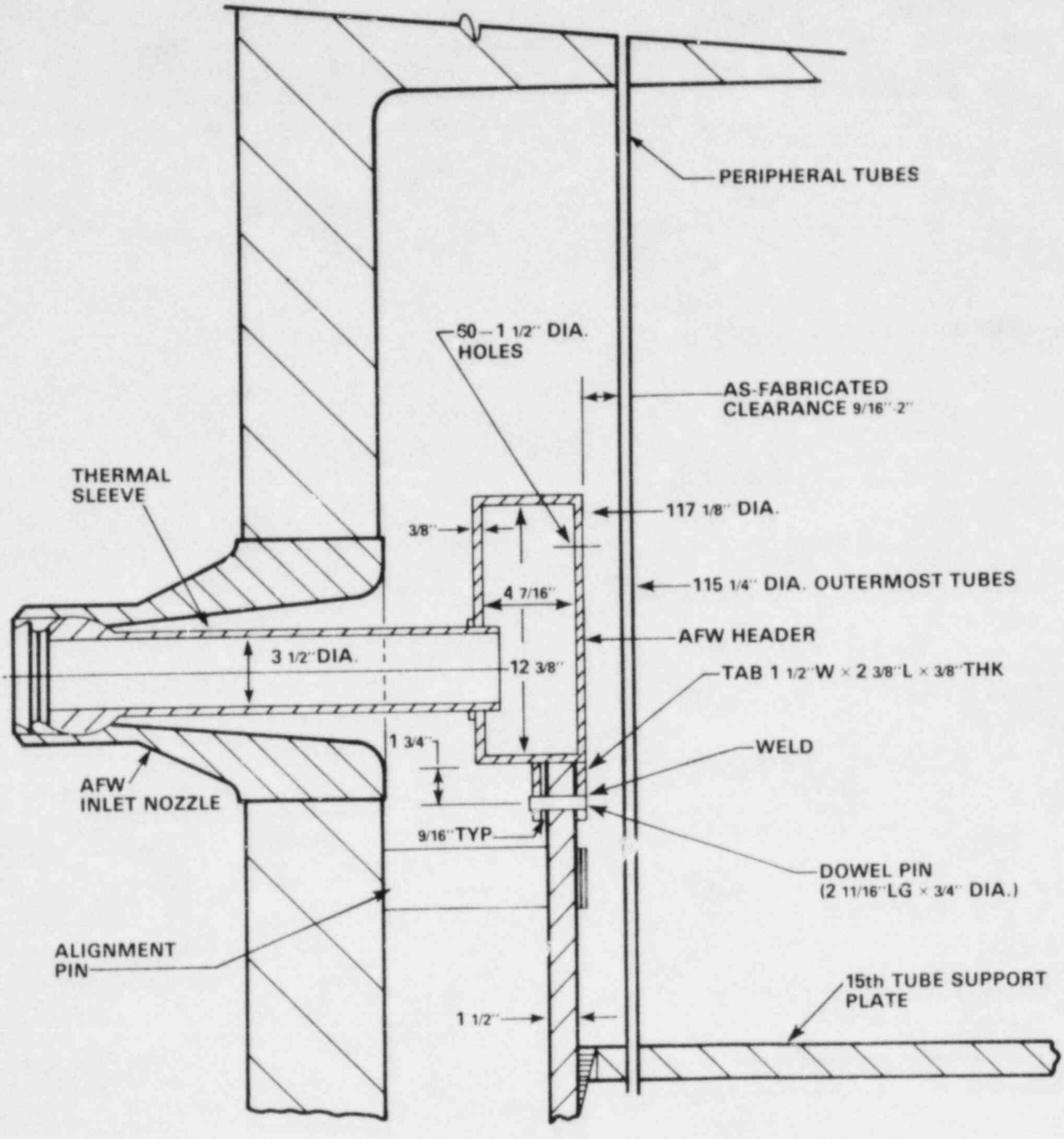

Figure 1. Internal AFW Header Design (Longitudinal Section) for Davis - Besse Unit 1

1-1/2-inch-diameter flow holes near the top of the inner header wall. The flow holes are equally spaced around the circumference. There are eight 1/4-inch-diameter drain holes near the bottom of the inner vertical wall. The auxiliary feedwater system piping connects to the AFW nozzle outside each steam generator. During power operation, the internal AFW header, thermal sleeve, and a portion of the horizontal piping are filled with dry superheated steam.

Description of Problem

The following description of the problem was extracted from a special report submitted by the Toledo Edison Company (Ref. B-12).

In April 1981, S/G tube leakage was experienced at the Davis-Besse Unit 1 station. An eddy current (EC) inspection determined that two adjacent peripheral tubes were leaking. The elevation and circumferential location of the tube leaks were aligned with the location of a header bracket pin. An expanded EC inspection carried out in this generator in the areas near the other dowel pins identified one additional tube diameter reduction (ding) indication which could be correlated to a dowel pin location.

In May 1981 tube leakage at Rancho Seco Unit 1 was identified. Although the leaking tube was adjacent to the inspection lane and not related to the header, an EC inspection was performed at all dowel pin locations. The inspection recorded dings in tubes at five of the eight dowel pin locations.

In February 1982, a leaking tube at the bundle periphery was identified at Oconee Unit 3. An EC inspection performed at four of the eight dowel pin locations recorded no tube indications.

As a result of these indications, more EC inspections of the peripheral tubes in the S/G at Davis-Besse Unit 1 were planned for their 1982 refueling outage. As a result of these inspections, visual examinations of the internal headers were made to check for loose dowel pins in the brackets attaching the internal header to the steam generator shroud. It was during this inspection that the header and bracket damage were first detected. The results of this inspection led to the inspections at Rancho Seco Unit 1 and at Oconee Unit 3. With one exception, the inspection results from all three plants were generally similar. The outer vertical wall of the header was distorted inward toward the center of the S/G, the support brackets were bent or damaged, and the dowel pins were either out of position or missing. The exception was the presence of three holes in the plates of the headers of steam generator "B" at Oconee Unit 3. Also, cracks were found in the corner welds of the headers at Oconee Unit 3 and Rancho Seco Unit 1.

Eddy current inspection at Davis-Besse Unit 1 showed peripheral tubes in the two S/Gs had indications which were interpreted to show contact with the internal header assembly at some point in time. Of these 24 tubes, 7 tubes had outside diameter (OD) indications, and 17 tubes had ding indications. Three of the OD indications exceeded the technical specification plugging limits of 40% throughwall. Further analysis indicated that the header was very near tubes around one axis and showed that there was only slightly more clearance at all other locations.

A visual inspection of the internal header at Davis-Besse Unit 1, followed by a 360° remote video inspection, showed that the outer wall (shellside) of the header was distorted inward (concave) as much as 4-1/2 inches. In addition, the inner vertical wall was noted to be bent inward in some locations. In one S/G, the thermal sleeve was disengaged from the inlet hole of the header and was offset from the center of that opening. It was also noted that certain header support brackets were bent, the bottom ligament torn out or were broken off, and that there was evidence of wear and/or distress on dowel pins and brackets. Dowel pins were missing at the majority of the eight bracket locations in each of the S/Gs. All brackets and all but one dowel pin have been located and retrieved.

Probable Consequences

There were a number of safety concerns associated with the design problem. There was the potential for tube rupture due to the interaction of the damaged headers with the peripheral tubes. There was the potential for damage to the S/G tubes and various primary system components from loose parts. In addition, there was the potential for degraded flow of the auxiliary feedwater which could have led to an inadequate heat sink for the reactor under certain off-normal conditions. AFW flow might also be diverted to the steam lines. However, during the operational history of the plants, no obvious adverse effects were noted during many actuations of the AFW systems.

Cause

It has been concluded that the most likely cause was rapid condensation-induced high pressure differential. Stress calculations performed by the licensees concluded that a pressure differential in excess of 200 psi is required to cause collapse of the internal headers. The design of the internal AFW header was such that during operation of the plant, prior to AFW actuation, the header would be filled with steam. When AFW was actuated, there would be a sufficient flow of subcooled water which would result in a rapid condensation of trapped steam, with attendant depressurization, inside the header. It is also concluded that the rapid flooding of the header with cold AFW when the steam generator was at operating conditions produced high thermal stress which could contribute to the distortion. S/G tube damage occurred due to interactions of the header, brackets, and dowel pins with the tubes.

The apparent cause of the above problems was a design deficiency in plants which utilize the internal AFW header design.

Corrective Actions

The auxiliary feedwater systems design and the associated operational experience were reviewed by B&W, a B&W Regulatory Response Group, and the NRC. The corrective actions consisted of (1) stabilizing the existing internal feedwater header to preclude further interaction with the steam generator tubes and discontinuing its use for flow distribution; (2) the installation of an external

AFW header with six injection nozzles (eight at Davis-Besse) to provide AFW flow, similar to the AFW ring header design which exists at five other B&W plants; (3) removal of loose parts that can be located (those that cannot be found must be shown to not pose a safety-related problem); and (4) the use of surveillance programs to assure no degradation of the systems and components involved. The NRC has completed safety evaluations for the restart of Rancho Seco Unit 1 and Davis-Besse Unit 1 (Refs. B-13, B-14) and will complete a safety evaluation for the restart of Oconee Unit 3.

Weld Cracking in Steam Generator Shells

On March 27, 1982, while Indian Point Unit 3 was in cold shutdown during a refueling outage, the Power Authority of the State of New York (the licensee) observed a leak in the girth weld between the upper shell and the transition cone of one of the plant's steam generators (Westinghouse designed). Inspection showed an oval-shaped through hole approximately 5/8 inch by 1/8 inch. Subsequent ultrasonic examinations of the corresponding weld in all four steam generators revealed that each had extensive cracking. There were an average of 170 cracks indicated per steam generator, with an average depth of 3/4 inch, a maximum indicated depth of about 1-1/2 inches, and a length of 2 to 4 inches. About 40% of the cracks were reported to be in weld metal.

Preliminary information indicates that the cracks were caused by corrosion fatigue probably accelerated by aggressive water chemistry and/or existing flaws from fabrication. Determining the exact interrelationships among these aspects will require extensive evaluation.

The significant factors that make this weld different from others in Westinghouse steam generators are: (1) this is a difficult final-closure weld to make, (2) this weld location had a local post-weld heat treatment rather than a furnace treatment, (3) this weld location is near the normal water line, and (4) this weld location is also near the feedwater ring and therefore may be subjected to thermal cycling. No other reportable indications of cracking have been found by the licensee in other welds in the steam generators of Indian Point Unit 3.

This plant has had a long history of condenser leakage problems, resulting in a small continuing inleakage of impurities into this unit even when major condenser leaks have not been identified. Constituents present in the sludge indicate that oxygen control in the feedwater/steam generator train may have been poor for a considerable length of time, because the licensee minimized the use of hydrazine due to environmental concerns. In January 1981, a turbine blade failed and fragments entered the condenser, causing a massive inleakage of chlorides that reached 325 parts per million in the steam-generator blowdown. (The cooling water from the Hudson River is brackish.)

The structural and operational features of Indian Point Unit 3 are common to many other Westinghouse units, and, therefore, the problem may be generic. Also, it is not known whether the problem is applicable to designs by Babcock & Wilcox or Combustion Engineering. Evaluation of these possibilities continues.

On the basis of this incident at Indian Point Unit 3, a similar occurrence at a foreign plant, and the high temperatures in the areas of concern, there should be a leak before failure and thus a warning before a significant event occurs.

The problem remains under active review. The specific cause is under investigation. Westinghouse has recommended inspections to 15 plants. The NRC must be assured that the problem and solution have been adequately addressed before Indian Point Unit 3 will be allowed to restart.

Internal Corrosion of Steam Generator Tubes

On November 21, 1981, General Public Utilities (the licensee) determined that the B&W designed "B" Once Through Steam Generator (OTSG) for Three Mile Island Unit 1 had a primary to secondary leak. The plant has been in a shutdown condition ever since the March 28, 1979 accident at Three Mile Island Unit 2. The tube degradation was determined to be due to intergranular stress corrosion, but was different from that typically encountered in other PWR plants in that the corrosion proceeded from the inside (primary side) of the tubes and worked outward - indicating that a corrosion agent or agents were in the primary coolant water.

The corrosion resulted in the formation of circumferential intergranular cracks. Approximately 98% of the cracks occurred in a length of 2 to 3 inches at the upper end of the tubes within the upper tubesheet near the roll transition area and the heat-affected zone of the upper seal weld. The number of tubes with defects is estimated at 16,000 to 20,000 of the total of 31,000.

Metallographic analyses by the licensee of portions of 19 removed tubes confirmed that sulfur in reduced forms was the aggressive agent causing the corrosive attack. The primary source of the sulfur was thiosulfate from the reactor building spray system, which entered the primary system at various times in 1981. The thiosulfate leaked past shut isolation valves in the spray system and entered the reactor coolant system during testing involving cross connections of various systems. (Sodium thiosulfate was required to be available for injection into spray water to capture iodine from air and water in the containment building in the event of an accident in the containment building resulting in the release of radioactive iodine.)

The licensee believes that the corrosive attack occurred at the end of or shortly after plant cooldown in September 1981 following hot functional testing, when conditions of susceptible material, aggressive chemical environment, and high stress existed. The attack was rapid, occurred primarily in the region where the tubes were exposed to air when the system water level was lowered, and probably terminated when the concentration of the aggressive sulfur species was reduced. Sulfur levels at the time of the corrosion are believed to have been on the order of several parts per million.

The thiosulfate tank was drained and flushed. Sulfur levels in the primary system then were less than 0.1 part per million. There is no evidence that the corrosion has continued since it was first discovered in late November 1981.

Most of the defects were in the upper 6 inches of the tubes. The licensee decided to perform an explosive expansion of the top several inches of each tube within the 24-inch thickness of the upper tubesheet (UTS) thereby closing the crevice area between the sheet and the tube and thus establishing a seal between the primary and secondary fluid. Qualification testing involving axial load testing, thermal cycling, leak testing, and pullout load testing is under way at various laboratories. NRR staff, with the aid of consultants, reviewed each of the major areas of the repair program. The licensee has completed preliminary qualification testing of the explosive expansion repair technique to be used. Following recommendations by B&W and Foster Wheeler, the licensee decided to expand the top 17 inches of all tubes within the 24-inch UTS and establish a 6-inch sealed area free of defects which will be the load carrying seal. By this method, all tubes with defects within the top 11 inches of the UTS can be saved. Consideration is being given to saving some additional tubes with defects below 11 inches by using a 22-inch expansion on selected tubes. It has also been determined that each expansion will require two detonations; one to expand the tube and the other to tightly seal the tube against the UTS.

The final qualification program consisting of performing various tests (including leak testing, axial pullout load testing, induced strain tests, thermal cycling tests, etc.) has commenced on a number of 10 expanded tube-in-tubesheet blocks. A full scale test on an OTSG at Mount Vernon, Indiana was conducted during August 1982. During this test, which was witnessed by the staff, a significant number of tubes were expanded using the prototypical process which will be used in the OTSGs at TMI-1. Expansions of tubes in the OTSGs commenced in November 1982. Tubes which cannot be saved by the repair technique described above will be plugged, removing them from service.

With respect to the potential that the corrosion attack may have affected other reactor coolant system (RCS) materials, the licensee outlined a detailed inspection program. All RCS materials were classified as to their corrosion susceptibility and an inspection plan developed which involves various non-destructive techniques as well as some destructive laboratory testing. The inspection program was initiated in parallel with the steam generator tube inspection, testing, and repair program. Any components found that would not be acceptable for use will have to be either repaired or replaced. Completion of RCS materials examination and testing indicated no corrosion attack of the RCS materials.

On May 12, 1982, the NRC issued Inspection and Enforcement Information Notice No. 82-14 to inform licensees of the corrosion problem (Ref. B-15).

Loose Parts in Steam Generators

Numerous instances of loose parts and/or foreign objects in steam generators have been discovered since November 1981 (See Table 1). In some cases, considerable damage was done to the steam generator tubes. Many of the cases were due to design deficiencies or inadequate maintenance/repair activities. The problem of loose parts, together with possible corrective/preventive actions, are under review.

Table 1 Steam generator loose parts since November 1981

Plant	Date	Licensee	NSSS*	Comments
Ginna Unit 1	1/82	Rochester Gas & Electric Corp.	Westinghouse	Problem described in AO 82-4 in NUREG-0090, Vol. 5, No. 1
Zion Unit 1	2/82	Commonwealth Edison Co.	Westinghouse	Fragments of a primary system nozzle cover found in S/Gs 1B and 1D (primary side). Some damage resulted to some S/G 1D tube ends.
Davis-Besse Unit 1	4/82	Toledo Edison Co.	B&W	Problem described previously in this report.
Rancho Seco Unit 1	4/82	Sacramento Municipal Utility District	B&W	Problem described previously in this report.
Oconee Unit 3	4/82	Duke Power Co.	B&W	Problem described previously in this report.
North Anna Unit 1	5/82	Virginia Electric & Power Co.	Westinghouse	Problem described in Appendix C of this report.
San Onofre Unit 1	5/82	Southern California Edison Co.	Westinghouse	Loose parts found in S/Gs A and B.
Cook Unit 1	7/82	Indiana & Michigan Electric Co.	Westinghouse	Loose parts found in S/Gs 11, 12, and 13.
Turkey Point Unit 4	7/82	Florida Power and Light Co.	Westinghouse	Loose parts found in S/Gs A, B, and C.

*Nuclear steam system supplier (and vendor of steam generator).

Steam Generator Tube Leaks

Several instances of steam generator tube leaks have occurred since November 1981. These are summarized in Table 2. Most were caused by mechanisms discussed previously under "Experience as of November 1981." Some leaks, such as at San Onofre Unit 1, Calvert Cliffs Unit 1, and Robinson Unit 2, were most likely caused by inadequate previous maintenance/repair activities. The tube leaks described in Table 2 are in addition to those reported in the sections above, e.g., Davis-Besse Unit 1, Ginna Unit 1.

Table 2 Steam generator tube leakage since November 1981

Plant	Date	Licensee	NSSS*
Oconee Unit 1	2/82	Duke Power Co.	B&W
Oconee Unit 3	2/82	Duke Power Co.	B&W
Zion Unit 1	3/82	Commonwealth Edison Co.	Westinghouse
Indian Point Unit 3	3/82	Power Authority of State of New York	Westinghouse
Millstone Unit 2	3/82	Northeast Nuclear Energy Co.	Combustion Engineering
Palisades Unit 1	3/82	Consumers Power Co.	Combustion Engineering
Oconee Unit 1	3/82	Duke Power Co.	B&W
Point Beach Unit 1	3/82	Wisconsin Electric Power Co.	Westinghouse
San Onofre Unit 1	3/82	Southern California Edison Co.	Westinghouse
Arkansas Unit 1	5/82	Arkansas Power and Light Co.	B&W
Calvert Cliffs Unit 1	5/82	Baltimore Gas and Electric Co.	Combustion Engineering
Yankee-Rowe Unit 1	6/82	Yankee Atomic Electric Co.	Westinghouse
Cook Unit 2	7/82	Indiana & Michigan Electric Co.	Westinghouse
Robinson Unit 2	7/82	Carolina Power & Light Co.	Westinghouse
Turkey Point Unit 4	7/82	Florida Power & Light Co.	Westinghouse
Beaver Valley Unit 1	8/82	Duquesne Light Co.	Westinghouse

*Nuclear steam system supplier (and vendor of steam generator).

Other Problems

1. Arkansas Unit 1 - The licensee (Arkansas Power & Light Company) has derated the unit because the pressure drop across the secondary side of the B&W-designed once-through steam generator had increased to the point where full flow could not be achieved. The problem is due to a crud (iron oxide) buildup, between support plates on the tubes, in the secondary side of the steam generator. Other B&W-designed plants may also possibly be

subject to the problem. B&W has the problem and possible corrective actions under review.

2. Maine Yankee Unit 1 - On March 10, 1982, the licensee (Maine Yankee Atomic Power Company) reported that during plant shutdown for maintenance, 6 of 20 steel studs failed during routine disassembly of one of the #2 Combustion Engineering-designed steam generator primary side manways. Ultrasonic inspection identified four more studs that were cracked. The studs exhibited evidence of surface corrosion attack, possibly as a result of an interaction associated with stud preload, lubricant, Furmanite, and primary coolant leakage environment. All 20 studs were replaced. The NRC requested Combustion Engineering to review the problem. NRC Inspection and Enforcement Notice No. 82-06 was issued to licensees on March 12, 1982 (Ref. B-16) to alert them of the event. This was followed on June 2, 1982 by NRC Inspection and Enforcement Bulletin No. 82-02 (Ref. B-17) which treated the problem as a generic issue. The Bulletin mentions that preliminary analysis by Combustion Engineering indicated that the failure mode was stress-corrosion cracking. The Bulletin required certain actions involving maintenance procedures, selection of materials, and information to be submitted for NRC review. The problem, and possible generic aspects, remain under review by Combustion Engineering, the licensees, and the NRC.

3. McGuire Unit 1 - As discussed above, McGuire Unit 1 (operated by Duke Power Company) is the only domestic operating plant utilizing the new Westinghouse-designed Model D steam generators. Inspections have shown tube wear due to tube vibration against the tube baffle plates similar to that found at Ringhals Unit 3 in Sweden and Almaraz Unit 1 in Spain, both of which use similar Model D steam generators. McGuire Unit 1 is restricted to not go over 50% power for any substantial period of time pending resolution of possible corrective actions. A possible fix is being tested in Sweden. Possible fixes are being actively pursued and/or reviewed by foreign governments, Westinghouse, the Westinghouse Owners Group of affected licensees, and the NRC. The affected plants include not only McGuire Unit 1 but also plants still under construction which plan to use steam generators based on the Model D version.

Further reports regarding unique operating experience or problems will be made as appropriate. This item is considered closed for purposes of this report regarding routine problems previously discussed in this series of reports.

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The following abnormal occurrence was originally reported in NUREG-0090-10, "Report to Congress on Abnormal Occurrences: October - December 1977," and updated in subsequent reports in this series, i.e., NUREG-0090; Vol. 1, No. 1; Vol. 1, No. 2; Vol. 2, No. 2; Vol. 3, No. 2; and Vol. 4, No. 2. It is further updated as follows:

77-9 Environmental Qualification of Safety-Related Electrical Equipment Inside Containment

The environmental qualification of electrical equipment for 71 operating reactors is being evaluated by the NRC staff and its contractors. Sufficient information has been supplied by all 71 licensees to enable the contractors to prepare a Technical Evaluation Report for each operating reactor. A total of 33 Technical Evaluation Reports have been issued. Based on the findings presented in the Technical Evaluation Reports, the NRC staff will prepare a Safety Evaluation Report for each operating reactor. It is anticipated that all Safety Evaluation Reports will be completed and issued to the licensees by mid-1983. Discussions among the licensees, the NRC staff, and its contractors will follow to resolve deficiencies identified in the Safety Evaluation Reports.

With regard to the environmental qualifications for operating license applications, there are 37 plants under review. Ten Safety Evaluation Reports have been issued. The remaining environmental qualification reviews of the operating license applications are at various stages of completion.

As previously reported, the Nuclear Regulatory Commission issued an Order on May 23, 1980 that required, by no later than June 30, 1982, all safety-related electrical equipment in all operating plants be qualified. The NRC plans to issue early in 1983 a final rule in regard to environmental qualification of electrical equipment important to safety for nuclear power plants. The final rule, when issued, will be applicable to all nuclear power plant licensees.

Since the Commission was unable to promulgate the final rule by June 30, 1982, and because licensees should not be placed in jeopardy of enforcement action pending promulgation of a revised schedule for implementation of equipment qualification requirements, the Commission issued an interim rule on June 30, 1982 in the Federal Register (Ref. B-18); the interim rule suspended the June 30, 1982 deadline for the licensees, imposed by the Commission Order of May 23, 1980, pending publication of the final rule. Previously, the NRC had evaluated each operating plant licensee's justification for continued operation; based on these analyses, the Commission determined that continued operation of these plants pending completion of the equipment qualification program will not present undue risk to the public health or safety.

The NRC also plans to issue later this year a revised Regulatory Guide which will describe procedures that would be acceptable to the NRC staff for complying with environmental qualification of safety-related electrical equipment requirments. The availability of a draft of the Regulatory Guide, together with a solicitation of public comments, was announced on February 22, 1982 in the Federal Register (Ref. B-19). Comments were received and are being evaluated.

Further reports will be made as appropriate.

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

79-3 Nuclear Accident at Three Mile Island

Reactor Building Entries

The tasks planned for the reactor building (RB) entry on April 14, 1982 had to be rescheduled when the first person to enter the building, a health physics technician, experienced problems with the airlock door mechanisms while exiting the building. The technician first attempted to open the doors in the normally used personnel airlock, and then the doors in the equipment hatch airlock. The equipment hatch airlock is designated as the alternate RB egress route. Both airlocks appeared inoperable. The technician eventually exited the RB through the normally used airlock by manually defeating a differential pressure interlock which had malfunctioned and disabled the door opening sequence. Following the entry, a leak test of the airlock door seals indicated that the equipment hatch door seals were leaking excessively (the leak path is into the reactor building since the latter is maintained at a negative pressure). A second entry was made late the same day to repair the equipment hatch airlock seals.

During the RB entry on April 22, 1982, a suction hose was installed into the incore instrumentation trough on the 282 ft. elevation floor. The trough extends below the general floor level and was chosen as the optimum suction location to remove the remaining water from the RB basement. Technicians also removed the reactor building dome radiation monitor, HPR-214. The dome monitor was shipped to an offsite laboratory for analysis.

The jet pump installation into the incore instrumentation trough to remove the remaining six inches of water from the RB basement was completed during the entry on April 28, 1982. The pump was energized from outside the RB on April 29, 1982 and began pumping the estimated 40,000 gallons (30,000 gallons of original sump water plus 10,000 gallons of water from the decontamination experiment) of water from the RB basement to the submerged demineralizer system (SDS) feed tanks at approximately 10 gpm. On May 2, 1982, the pumping operation was stopped when filters on the inlet line to the SDS feed tanks reached 40 psi of differential pressure (plugged filter indication) and pump flow decreased to 2 gpm. Approximately 36,000 gallons of water were pumped from the sump before the high differential pressure was reached, decreasing the water level from approximately six inches to one inch. The water level in the RB basement is expected to increase gradually from reactor coolant system leakage (about 100 gallons per day).

Additional work inside the RB on April 28, 1982, included the testing of six RB smoke detectors. The smoke detectors are part of the RB fire protection system and are designed to activate an alarm in the control room and automatically to secure portions of the RB ventilation system. None of the six detectors performed their designed functions during the test.

A radiation survey in the RB on May 5, 1982, following the removal of the 36,000 gallons of water from the basement, indicated that the water removal did not significantly reduce the general radiation levels on the 305 ft. elevation.

Areas on the 305 ft. elevation which are not well shielded from the basement (stairwells and metal gratings) did show a substantial gamma dose rate decrease following the water removal.

During the entry on May 10, 1982, the time was spent assessing the repair requirements of the polar crane.

During the entry of May 26, 1982, radiation surveys and valve inspections were performed inside the "D-rings" (primary system radiation barriers) to support the planned depressurization and partial draining of the reactor coolant system (RCS). Surveys and inspections were made at high point vent valves which will be used initially to vent the primary system components and then to inert the primary system with nitrogen as the RCS water is drained to a level which will permit opening a control rod drive mechanism (CRDM). The CRDM disassembly and CRD lead screw removal is sequenced to support a closed circuit television inspection of the reactor vessel upper internals in July 1982. The closed circuit television camera will be lowered into the reactor vessel through the disassembled CRDM.

A remote radiation survey was also performed inside the B "D-ring". This was the first survey inside the B "D-ring". The survey results indicated that the radiation levels in the B "D-ring" were substantially higher than in the A "D-ring" (the pressurizer is in the A "D-ring").

Prior to exiting the RB, the B "D-ring" survey team lowered a remote survey detector to the floor of the RB basement in the area below the open stairwell. Radiation levels increased gradually from 2 R/hr to 45 R/hr as the detector was lowered from the 305 ft. elevation to the 282 ft. elevation. Residue on the tip of the detector (the detector physically contacted the 282 ft. elevation floor) looked like wet sand.

Following fire protection system repairs, smoke detectors inside the RB were also tested. The fire detectors activated fire protection circuitry outside the RB, however, the fire protection system was still not fully operable (some local alarms did not activate and a ventilation damper failed to close). The problems appeared to be associated with circuitry outside the RB.

The in-containment nitrogen system was tested during the RB entry on June 10, 1982 and found to be operable. The nitrogen system will be used to inert the primary system components when the reactor coolant water level is lowered in preparation for the control rod drive lead screw removal and closed circuit television inspection of the vessel upper internals.

During the RB entry on June 17, 1982, a radiation survey was made in the open stairway going down to the 282 ft. elevation. A health physics technician descended the stairs from the 305 ft. elevation to a landing approximately eight feet above the basement floor (282 ft. elevation). Radiation levels at the landing were 12 R/hr gamma and 35 rad/hr beta. A radiation survey approximately three feet above the basement floor indicated that the gamma field was 24 R/hr. Based on a visual observation, the basement floor was described as completely covered with a brown liquid material. The depth of the material was difficult to estimate, and may range from a fraction of an inch to several inches.

On June 22, 1982, technicians placed "TLD trees" (strings of thermoluminescent dosimeters) below the 305 ft. elevation floor to map the radiation fields in the basement. Additional work included electrical maintenance on the spider lift and installation of a portable power supply on the 347 ft. elevation.

On June 23, 1982, a radiochemist and a health physics technician successfully descended the open stairwell to within a few feet of the 282 ft. basement elevation and obtained a scrape sludge sample, radiation readings, and photographs of the RB basement floor. Other technicians completed water flushes from the 305 ft. elevation of the containment walls below and attached acoustic monitors (microphones) to the reactor vessel in preparation for the APSR insertion test (described below).

Submerged Demineralization System (SDS)

The SDS processed approximately 229,300 gallons of reactor coolant bleed tank water during the second calendar quarter of 1982. Of that, approximately 15,000 gallons was from the containment building sump, 196,400 gallons was reactor coolant system water, and 17,900 was miscellaneous flush water.

Reactor Coolant System Cleanup

The first feed and bleed cycle for reactor coolant system cleanup began on May 17, 1982. A total of 196,400 gallons were cycled through the SDS for cleanup during the second quarter.

Axial Power Shaping Rod (APSR) Movement

The APSRs are control rods which contain boron (a neutron adsorber) in the lower one quarter of their length. They were designed to be moved within the reactor core to adjust the shape of the neutron flux when the reactor is at power. They do not serve a shutdown function and remain stationary when the reactor is shut down. The APSRs at TMI-2 remained at approximately the 25% withdrawn position when the reactor shut down at the start of the March 28, 1979 accident.

The licensee planned, if possible, to fully insert the rods so that they may later be decoupled from the lead screw before reactor vessel head removal. If the APSRs could not be driven in, an alternate decoupling method would have to be developed prior to reactor vessel head removal. As part of the procedure for APSR movement, the licensee planned to conduct tests on the APSRs, which could be of help in assessing damage to the reactor core and internal components.

The APSR movement began June 23, 1982. An RB entry was made during which acoustic monitors (microphones) were installed on the APSR drive mechanisms to monitor noise during the rod movement test.

Subsequently, the first of eight APSRs (No. 62) was withdrawn 3/16 of an inch in 1/32 of an inch increments, in accordance with the prepared test procedure. The rod movement direction was reversed and then the rod was reinserted into the reactor core. However, increased resistance to rod insertion was observed when the APSR reached about 5.5% of its "full-in" position (about eight inches from the bottom). The APSR test was stopped at this point since the 30-minute elapsed time limit, based on heat buildup, had been reached for APSR motor energization.

The APSR test on No. 62 was resumed on June 24, 1982. However, no further inward movement was possible at the maximum permissible electrical current (14 amps) which could be applied to the drive mechanism. APSR testing was continued on three more rods (Nos. 63, 65, and 66). These rods, like rod No. 62, were first withdrawn 3/16 of an inch. Then their direction was reversed and attempts were made to drive them into their "full-in" position. Only rod No. 65 reached its approximate "full-in" position.

APSR testing was completed on the final four rods (Nos. 64, 67, 68, and 69) on June 25, 1982. One of these rods reached its approximate "full-in" position (No. 67); two of the rods (Nos. 64 and 69) could not be moved inward.

The "as left" rod positions are summarized as follows:

Rod No.	Core Location	"As Left" Position (See Note below)
62	F-4	5.5%
63	L-4	18.8%
64	N-6	25.0%
65	W-10	0.1%
66	L-12	4.2%
67	F-12	1.1%
68	D-10	22.9%
69	D-6	26.1%

Note: The distance from full rod removal (100%) to full rod insertion (0%) is 144 inches. A 5.5% "as-left" position means a rod is approximately eight inches from the reactor core bottom.

Advisory Panel

On April 22, 1982, the Three Mile Island Advisory Panel held a meeting in Harrisburg, Pennsylvania. Various representatives from GPU Nuclear, NRC, DOE, and EPA gave status reports on (1) funding proposals in Congress, (2) the recent reactor building decontamination experiment, (3) accident water processing, (4) reactor coolant system processing, and (5) DOE's agreement to remove purification system demineralizer vessels and filters.

Extensive discussion took place on the subject of tritium, which has been showing up in groundwater adjacent to TMI-2 buildings as a result of a leak in the area of the Borated Water Storage Tank during January 1982. The relative levels of radioactivity in the area have been minimal (approximately one curie of tritium). However, the amount of water leakage, originally thought to be 50-60 gallons, has been determined to be closer to 2,000-3,000 gallons. Mr. William Kirk (EPA) reported undetectable levels of radionuclides in five EPA monitored wells just off the island.

Chairman Minnich also offered three proposals which were passed by the Panel:

1. Send a letter to the legislative leadership of the Commonwealth of Pennsylvania endorsing that the $5 million for the cleanup proposed by Governor Thornburgh in the FY83 budget, remain intact through the budget process.

2. Send a similar letter to Congress reinforcing the need and the Panel's support of $27 million for TMI cleanup proposed for the FY83 DOE budget.

3. Send a letter to the Pennsylvania PUC suggesting the $23 million currently slated for amortization of TMI-2 capital costs pursuant to deferred energy rate charge, be diverted to the cleanup. Monies slated for rate reduction relief would remain intact. Chairman Minnich felt such a move would be a positive sign to Congress of Pennsylvania's commitment to the cleanup.

<u>NRC-DOE Memorandum of Understanding on TMI Wastes</u>

In accordance with the USNRC and USDOE Memorandum of Understanding, the first SDS waste vessel (D-10015) was shipped from TMI to Richland, Washington on May 21, 1982. This ten cubic foot SDS vessel, which was used to process waste from the reactor coolant bleed tanks, contained approximately 13,000 curies of total radioactive material and was shipped in a special type B (designed for accidents) cask. The DOE, who took possession of this waste material at TMI, will sponsor research and development glass vitrification (solidification) testing at the Hanford, Washington facility. The NRC and DOT inspected the shipping container and transport vehicle to insure conformance with applicable Federal regulations. The shipment arrived at the DOE facility on May 24, 1982.

Further reports will be made as appropriate.

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80-7 <u>Loss of Salt Water Cooling System</u>

This item was previously reported and closed out in NUREG-0090, Vol. 3, No. 3 ("Report to Congress on Abnormal Occurrences, July-September 1980") with the title of "Failure of Salt Water Cooling System." It is being reopened and retitled to report an incident on May 13, 1982 involving the same system and the same plant (San Onofre Unit 1) as the March 10, 1980 incident reported in NUREG-0090, Vol. 3, No. 3.

The plant was shut down on February 27, 1982 for various inspections, modifications, and maintenance items. During this period, work was scheduled on May 13, 1982 for the removal of the south salt water cooling pump (SWCP) for maintenance.

The reactor was in cold shutdown. The reactor coolant piping had been drained for steam generator tube inspection. The upper Component Cooling Water Heat Exchanger was removed from service and open, and the south SWCP motor had been removed earlier in the day. The auxiliary SWCP circuit breaker was removed for maintenance and the pump's flow path was isolated. This isolation was needed for south SWCP maintenance and upper component cooling water heat exchanger maintenance. The north SWCP was operating and removing reactor decay heat from the component cooling water system via the lower component cooling water heat exchanger. The north and south screen wash pumps were operable in their normal alignment. (The screen wash pumps can be manually cross-connected to provide backup to the salt water cooling pumps, but they are of a lower capacity and are not qualified as safety-related equipment.)

At approximately 8:00 a.m., after removing the south SWCP motor and the nuts attaching the pump to its pedestal, a crane removed the pump. Ocean water immediately entered the intake structure through the resulting hole in sufficient quantity to prevent personnel from reseating the pump on its foundation. Flooding continued until 8:42 a.m. when the level in the intake structure rose to sea level, approximately five feet above the floor of the intake structure. At this time maintenance personnel were able to partially remount the south SWCP and begin to slowly reduce the water level in the intake structure using a portable pump.

All salt water cooling was lost from 8:18 a.m. until 8:42 a.m. when operators completed manual valve alignment to allow the north screen wash pump to supply salt water cooling to the lower component cooling water heat exchanger. At 8:29 a.m. a spare breaker for the auxiliary SWCP was installed and the auxiliary SWCP was considered available if needed. Licensee personnel, however, decided not to use the auxiliary SWCP because the north screen wash pump was operating and adequately removing decay heat from the reactor.

Salt water flow was lost from the north SWCP because the control operator secured the north SWCP when the pump amperage and the pump discharge valve began to cycle erratically. When flooding ceased, the intake structure water level was approximately 2 inches below the pump motor vents for the north salt water and the south screen wash pumps, and approximately 18 inches below the north screen wash pump motor vents.

The north SWCP was returned to service at 1:14 p.m. but failed at 2:34 p.m. because of a failed-shut discharge valve. Salt water cooling was briefly interrupted, until the north screen wash pump was aligned to supply salt water cooling. This abnormal alignment was maintained until 6:45 p.m. when the north SWCP was returned to service. Apparently, the pump discharge valve failed closed due to residual moisture in the pressure switch and melted insulation in an associated time delay relay.

During the 24-minute period when no salt water cooling was available, reactor coolant core outlet temperature did not rise perceptibly, reactor coolant inlet temperature rose 1.5°F, and component cooling water outlet temperature rose 15°F.

The procedure specified for this work had one precaution to prevent flooding: "To prevent flooding, remove and install the pump at the low tide only." This precaution was inadequate to prevent flooding because no specific tide level was specified.

Maintenance personnel had used a tide chart for Los Angeles (Outer Harbor), 45 miles northwest of San Onofre, to estimate the time of low tide as 8:06 a.m. on May 13, 1982. The maintenance foreman and watch engineer estimated that the tide level would be approximately two feet up the three-foot-tall pedestals of the saltwater cooling pumps. In fact, the water level reached was three feet higher than this. This technical error resulted in the flooding and is considered the principal cause for the event.

The NRC's and the licensee's investigation of this event identified several primary causes: inadequate communication between the operators and maintenance personnel, an error in estimating the low tide level, and inadequate maintenance procedures.

The licensee is upgrading the applicable procedures and has requalified maintenance and operations personnel in the proper implementation of equipment control. Formal checklists for equipment removal and restoration will be established for the salt water cooling system.

On June 16, 1982, the NRC Region V Office cited the licensee with a notice of violation for the failure to provide sufficient detail in the SWCP maintenance procedure to safely account for the effect of ocean tide conditions (Ref. B-20). The fact that the SWCPs were inoperable was not a violation since the license Technical Specifications do not require the salt water cooling system to remain operable while the reactor is in a cold shutdown condition.

Subsequently, on August 13 and 19, 1982, two more incidents involving this system have occurred. On August 13, while restoring the south SWCP to service from the May 13 event, with the north SWCP supplying flow to the lower component cooling water heat exchanger, the discharge valve on the south SWCP opened unexpectedly. This allowed much of the running north SWCP flow to flow in reverse through the idle south SWCP, bypassing the heat exchanger. This valve was immediately shut by an operator at the valve, restoring full flow to the heat exchanger. The brief reduction of the flow through the heat exchanger did not observably increase core temperatures. However, the incident was of concern because the pump discharge valves had been observed to be occasionally unreliable and erratic. This was first noted by the NRC during the March 10, 1980 event review, and more recently following reviews of January 18, February 1, and March 19, 1982 incidents.

On August 19, 1982, the north SWCP had to be removed from service due to a smoking lower motor bearing. At the time of this incident, the south SWCP remained out of service from the May 13, 1982 incident. The nonsafety-related auxiliary SWCP was started to maintain sufficient salt water flow to the upper heat exchanger. Temporary repairs to the north SWCP were made using a spare motor for this pump. During this period of several hours, the nonsafety-related auxiliary SWCP and screen wash pumps were used to provide salt water flow. Subsequent investigation of the pump motor determined that the inside of the motor was rusty, muddy, and oily, the lower motor bearing was wiped, and the pump upper bearing was excessively worn.

The losses and reductions of salt water cooling had no adverse effects on public health or safety. However, as shown in a case study report (Ref. B-21) prepared by the NRC's Office for Analysis and Evaluation of Operational Data for the event on March 10, 1980, a complete loss of the salt water cooling system during the early stages of RHR operation could lead to damage to some safety-related equipment within a few minutes. Such single failure vulnerability of cooling water systems is under review as part of the NRC's systematic evaluation program. In addition, the NRC's Region V is closely monitoring the licensee's engineering and human factors studies to improve the reliability of this system. In the interim, additional operator training on special operation precautions for this system is planned.

This incident is closed for the purposes of this report.

APPENDIX C

OTHER EVENTS OF INTEREST

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

1. Temporary Total Loss of High Head Safety Injection Capability

On February 12, 1982, while operating at 50% power, all three charging pumps of the McGuire Nuclear Station Unit 1 became inoperable, thereby resulting in the total loss (for a period of 38 minutes) of the functions associated with these pumps, including high head safety injection for system pressure above 1500 psig. McGuire Unit 1 utilizes a Westinghouse-designed pressurized water reactor. The unit is operated by Duke Power Company (the licensee) and is located in Mecklenburg County, North Carolina.

McGuire Unit 1 utilizes three pumps, all with a common suction, to supply makeup flow (charging) to the reactor coolant system (RCS). One of these pumps is a reciprocating charging pump, commonly referred to as a positive displacement pump (PDP). The other two pumps are centrifugal charging pumps (CCPs). During normal operations, most of the reactor coolant is circulated through the RCS by the reactor coolant pumps. A portion of the reactor coolant flows from a pipe (letdown line) connected to one of the reactor coolant inlet pipes (cold leg) to the reactor vessel. This reactor coolant flows to the chemical and volume control system (CVCS). Flow then enters the common suction of the charging pumps. During normal operations, either the PDP or one of the CCPs is used for charging. From the pump discharge, the flow is split into two paths - one for reactor coolant pump seal injection and the other for charging flow through one of the cold legs of the reactor pressure vessel.

The two CCPs also serve as the motive force for the high head safety injection system by providing protection for steamline breaks or for small RCS breaks where the pressure remains above 1500 psig for an extended time period. If RCS pressure drops to about 1500 psig, the intermediate head safety injection system provides additional injection flow.

The system described above is typical of several Westinghouse-designed plants. The McGuire Unit 1 system, however, had an additional feature which was directly involved in the February 12, 1982 event, resulting in the loss of all three charging pumps. Due to the pulsating suction flow, characteristic of reciprocating pumps, the PDP was equipped with a suction dampener consisting of a vertical section of 12-inch pipe, containing water with a hydrogen gas cover providing overpressure. The water level is controlled by two solenoid valves which respectively supplies hydrogen gas when the water level is too high and vents off gas when the water level is too low. These valves can be controlled automatically by water level switches, or manually by switches mounted on a local panel.

On February 12, 1982, the plant was operating at 50% power with the PDP out of service for modification; charging flow was being provided by CCP-1A. During an attempt to fill and vent the PDP suction piping in preparation for returning the pump to service, the opening of the suction isolation valve to the PDP resulted in air and hydrogen entering the PDP pump suction piping and into the common suction of the two CCPs. Later investigation indicated that the dampener level control system had malfunctioned, resulting in a continuous supply of hydrogen to the dampener and a consequent displacement of water (later estimated to be about 50 cubic feet) by the gas. Control room personnel observed oscillation of the CCP-1A motor current and charging flow and thus switched to CCP-1B, shutting down CCP-1A. About 30 seconds later, similar indications for CCP-1B were observed. The pump was shut down and letdown was isolated.

With both CCPs inoperable, there was a temporary total loss of emergency core cooling capability above 1500 psig. The plant technical specifications require that the reactor be shut down within one hour if the CCPs cannot be restored to operable status within that time. Plant personnel were able to get pump 1B restarted and operating properly in about 38 minutes and subsequently letdown flow was reestablished.

During the event, reactor coolant inventory was decreasing due to normal reactor coolant pump seal leakage of about 12-15 gpm. No makeup capability existed since all three charging pumps were inoperable. A significant decrease in the reactor coolant system average temperature would have resulted in a further drop in pressurizer level (due to contraction of the coolant) although system pressure could have been maintained by pressurizer heaters until pressurizer level dropped below 17%; at this point, the heaters would automatically shut off to prevent their overheating. If the restoration of reactor coolant makeup had been delayed until after the pressurizer inventory was lost, or had a transient occurred forcing a loss of pressurizer inventory, a reactor shutdown would have occurred. With the loss of pressurizer level, loss of pressure control would occur. The lower limit of the pressure excursion would be determined by system hot spot saturation pressure. Reactor coolant temperature trends would be dependent on control of the heat transfer rates to the steam generator and on the core decay heat generation rate.

Had a reactor shutdown occurred because of low pressurizer pressure, an automatic safety injection signal would have started the intermediate head safety injection pumps to provide adequate core cooling.

The cause of this event is attributed to design inadequacies. Failures in the nonsafety-related suction damper for the PDP resulted in the loss of both CCPs. After the event, the licensee examined the dampener level control system. An empty reference leg in the system was found which would result in a continuous supply of hydrogen to the dampener; this would displace the water in the PDP suction piping and result in air and hydrogen flowing into the common suction of the two CCPs when the PDP suction isolation valve was opened. The cause of the reference leg being drained could not be determined; no leaks were found when the reference leg was refilled. Also contributing to the event were procedural/personnel errors. Had the preparations been adequate for returning the PDP to service, the event should not have occurred.

As corrective actions, the licensee isolated the PDP and suction dampener from the charging system. The licensee is evaluating the system design to determine what temporary and/or permanent changes are necessary to prevent recurrence of this type of event. The PDP will not be returned to service until such a temporary or permanent change is made.

A Confirmation of Action Letter was issued by the NRC to the licensee on March 25, 1982 (Ref. C-1) to confirm the requirement to isolate the suction dampener from the charging system, to evaluate system design changes to prevent recurrence, and to submit the evaluation to the Regional Office prior to returning the PDP and suction dampener to service. The NRC issued Inspection and Enforcement Information Notice No. 82-19 on June 18, 1982 to nuclear power reactor facilities to inform them of this event (Ref. C-2).

Based on the information to date, some of the concerns associated with the event were the licensee's lack of specific procedures to follow and the lack of a safety analysis for such an event. However, it should be noted that some Westinghouse-designed plants do not use the charging pumps for a safety injection at RCS pressure above 1500 psi. These plants use only an "intermediate" head system to mitigate small or large pipe breaks. The probability is very low for the simultaneous loss of all charging flow together with a small (or large) pipe break. The NRC's Office for Analysis and Evaluation of Operational Data is continuing to review the event for possible abnormal occurrence implications.

2. Reactor Fuel Degradation

On April 26, 1982, scheduled inspections of reactor fuel assemblies at the Trojan Nuclear Plant identified abnormal degradation of several 17 x 17 reactor fuel assemblies. The Trojan Nuclear Plant, operated by Portland General Electric Company (the licensee), utilizes a Westinghouse pressurized water reactor and is located in Columbia County, Oregon.

In late 1981, during Cycle 4 operation of the Trojan Nuclear Plant, higher than normal fission product and gross activity levels were detected in the reactor coolant. These levels were monitored carefully by the licensee and were observed to slowly increase (except during periods of plant shutdown and power reduction) until the facility began a planned refueling outage in April 1982. The observed coolant activity levels remained below the limits provided in the facility license.

The inspections performed by the licensee identified eight damaged peripheral assemblies by visual examination using an underwater TV camera. Nine other fuel failures, not obvious by visual examination, were detected by fuel sipping, a technique which checks for the release of fission products from the fuel assembly.

An investigation into the cause of the fuel failures by the licensee and Westinghouse (the fuel supplier) determined the failures were due to fuel rod vibration. Vibration of the fuel rods, all in peripheral locations in the core, was caused by water jetting through joints in the core baffle, a bolted steel assembly which surrounds the reactor core.

To prevent a recurrence of the fuel failures, the licensee has replaced a number of the fuel rods most subject to vibration (in peripheral assemblies) with solid stainless steel rods. Additional stiffener grids were installed on fuel assemblies adjacent to the baffle joints. These modifications have had a negligible effect on core performance; this was verified by testing during plant startup. During a future refueling outage, the licensee plans to modify the core baffle in a manner which will eliminate the baffle jetting phenomenon.

The observed fuel failures had no effect on public safety or the environment and did not result in radioactivity levels or effluent releases in excess of those allowed by the operating license. Since the particular fuel damage was not considered a major degradation, the incident was deemed not reportable as an abnormal occurrence.

NRC Inspection and Enforcement Information Notice No. 82-27 was issued on August 5, 1982 to licensees to inform them of this event (Ref. C-3).

3. <u>Control Rod Drive Guide Tube Support Pin Failures</u>

On May 17, 1982, Virginia Electric and Power Company began a refueling outage one week early due to detection by the Loose Parts Monitoring System of a possible foreign object in steam generator (S/G) "A" of North Anna Unit 1. The plant utilizes a Westinghouse-designed pressurized water reactor and is located in Louisa County, Virginia. During subsequent inspections, the lock nut of a control rod drive (CRD) guide tube support pin was found in S/G "A" and a smaller piece of material, identified also as part of a support pin, was found in S/G "C". About 75% of the tube ends sustained damage. The presence of loose parts in the two S/Gs had been established only 24 hours before the plant was shut down. No foreign objects were found in S/G "B" and its tubes were not damaged. The pins which caused the problem in Unit 1 had been replaced in Unit 2, prior to initial startup, by pins using a different heat treatment. The two failed pins in Unit 1 had come from two guide tubes (one failed pin per guide tube). All 61 guide tubes were removed. During the removal process, one pin failed in each of 17 guide tubes and both pins failed in 3 guide tubes. The licensee is replacing all 61 CRD guide tubes and will use support pin assemblies now being recommended by Westinghouse. While extensive damage occurred to the S/G "A" and "C" tube ends, the licensee (and Westinghouse) does not consider the damage to be significant such as to prevent testing of the S/Gs or to affect the performance.

Prior to the event at North Anna Unit 1, these failures had occurred only at foreign reactors (Japan and France). Pin failures have occurred with both Westinghouse and a foreign firm-supplied pins. The first failures were detected in early 1978 at a Japanese plant. The pins are used to align the bottom of the CRD guide tube assembly into the top of the upper core plate. Two support pins are bolted into the bottom plate of each lower guide tube and are inserted into the top of the upper core plate in a manner that provides lateral support while accommodating thermal expansion of the guide tube relative to the core plate. The pins are about 3½ inches long and have a diameter of

0.507 or 0.537 inch (depending upon reactor design). The pin assembly includes: (1) a bolt section to which a nut (sleeve) is threaded to anchor the pin to the guide tube, (2) a collet which rests against the guide tube, and (3) a leaf spring section - the leaf being somewhat like a clothespin. Pin material is Inconel X-750 which, depending on the manufacturer and the fabrication date, has been solution heat treated and age hardened at various temperatures and for various times.

Westinghouse analysis indicated the failures are caused by a combination of stress corrosion cracking (SCC) and high stresses on the shank and leaf spring section of the pins. Westinghouse recommends a revised manufacturing process to prevent SCC, combined with a lower torque on the lock nuts.

The consequences of pin failure for plants with the upper head injection (UHI) design was originally considered more acute than those for non-UHI plants. This concern was due to the potential for CRD misalignment in UHI plants upon pin failure. However, the domestic operating UHI plants now have support pins meeting the recommended material process standards and the pin body design has been revised to prevent control rod misalignment upon pin failure.

The problem of pin failure in non-UHI plants, the possible consequences, and the possible generic aspects of the problem remain under review by the NRC. In addition, the NRC is closely monitoring the licensee's efforts and will review the adequacy of any corrective action required prior to restart of North Anna Unit 1. Restart of the plant is not expected until at least late fall of 1982. NRC Inspection and Enforcement Information Notice No. 82-29 was issued on July 23, 1982 to notify licensees with Westinghouse-designed plants of the event (Ref. C-4).

The event at North Anna Unit 1, a non-UHI plant, resulted in no effect on public health or safety. The licensee took expeditious corrective actions, when possible foreign objects were detected in the S/Gs, before serious consequences occurred. However, the problem remains under review by the NRC.

4. Multiple Diesel Generator Failures

On June 2, 1982 the NRC was notified by Baltimore Gas and Electric Company (the licensee) that all three diesel generators were simultaneously inoperable from 7:05 am until 7:35 am and only one offsite power circuit was operable at the Calvert Cliffs Nuclear Power Station. The Calvert Cliffs Station utilizes two Combustion Engineering-designed pressurized water reactor plants and is located in Calvert County, Maryland.

On June 2, 1982, with Unit 1 shut down for refueling and Unit 2 at full power, a series of events occurred which resulted in no emergency power sources at the Calvert Cliffs Nuclear Power Station for a period of 11 minutes. The Unit 1 diesel generator (DG-11) was out of service for maintenance. Prior to removing a 500-kV electrical distribution bus from service to perform annual routine maintenance on a transformer, the other two of the three on-site emergency diesel generators (DG-12 and DG-21) were tested successfully. Following the performance test, the swing diesel (DG-12) was returned to standby status and the Unit 2 diesel (DG-21) continued to run fully loaded in parallel with the grid. The 500-kV transformer was then taken out of service, leaving only a single circuit capable of providing off-site power to the station. DG-21 tripped off (shut down) due to voltage regulator drift. DG-12 was then started and fully loaded to demonstrate its operability. At 7:05 am, DG-12 tripped as a consequence of an operator error while raising the main generator voltage in response to a request from the electrical load dispatcher. The DG trips were reset and both diesels restarted at 7:16 am. Following a 15-minute load test for operability, DG-12 was declared operable at 7:35 am and DG-21 operable at 8:00 am. This terminated the incident and satisfied the Action Statements of Technical Specifications.

The licensee did not initiate the Emergency Response Plan for the event since a reactor mode change was not made (the condition was corrected within the time period allowed by the plants's Technical Specifications). The licensee changed operating instructions to alert operators to monitor DG reactive load when adjusting system line voltage.

Both of the DG trips occurred during parallel operation of the diesel generators and involved the action of a loss-of-field protective relay. This relay is not active when the generator is in the emergency mode and therefore not paralleled with other electrical sources. The diesel generators would therefore have been available, after the lock-out relays were manually reset at the diesel, in event of loss of off-site power.

The safety implication of the simultaneous inoperability of three diesel generators bears on availability of emergency power in event of an accident. If such an accident had involved loss of offsite power, it would have resulted in a temporary loss of power at the facility until the diesel generators could be reset and manually restarted in their accident mode. The event did not result in any adverse effects on health and safety of the public or licensee personnel.

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* Available in NRC Public Document Room, 1717 H Street, NW., Washigton, 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, Virginia 22161. This series of reports is also available on a subscription basis from the NRC-GPO Sales Program.

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Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event which the Nuclear Regulatory Commission determines to be significant from the standpoint of public health or safety and requires a quarterly report of such events to be made to Congress. This report covers the period April 1 to June 30, 1982.

During the report period, there were no abnormal occurrences at the nuclear power plants licensed to operate. There were no abnormal occurrences for the other NRC licensees. The Agreement States reported no abnormal occurrences to the NRC.

The report also contains information updating some previously reported abnormal occurrences. Some of the updates have been given more generalized titles (as compared to their former more specific titles) to include some new events which are associated in some respects to previously reported abnormal occurrences. The items which have been retitled are discussed in Appendix B of the report.

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