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Vol. 6, No. 3

Report to Congress on Abnormal Occurrences

July - September 1983

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 July 1 to September 30, 1983.

The report states that for this report period, there were three abnormal occurrences at the nuclear power plants licensed by the NRC to operate. The first involved large diameter pipe cracking in boiling water reactors; the second involved an uncontrolled leakage of reactor coolant outside primary containment; and the third involved improper control rod manipulations. There were seven abnormal occurrences for the other NRC licensees. Three involved overexposures; two involved medical misadministrations; one involved widespread radiological contamination; and one involved willful violation of license and a material false statement to the NRC. There were no abnormal occurrences reported by the Agreement States.

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

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PREFACE

INTRODUCTION

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

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

The NRC has reviewed Licensee Event Reports, licensing and enforcement actions (e.g., notices of violations, civil penalties, license modifications, etc.), generic issues, significant inventory differences involving special nuclear material, and other categories of information available to the NRC. The NRC has determined that only those events, including those submitted by the Agreement States, described in this report meet the criteria for abnormal occurrence reporting. This report covers the period between July 1 to September 30, 1983.

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, information on reportable events received from NRC licensees is routinely sent to the NRC's more than 100 local public document rooms throughout the United States and to the NRC Public Document Room in Washington, D.C.

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

AGREEMENT STATES

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

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

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

FOREIGN INFORMATION

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

REPORT TO CONGRESS ON ABNORMAL OCCURRENCES

JULY - SEPTEMBER 1983

NUCLEAR POWER PLANTS

The NRC is reviewing events reported at the nuclear power plants licensed to operate during the third calendar quarter of 1983. As of the date of this report, the NRC had determined that the following were abnormal occurrences.

83-5 Large Diameter Pipe Cracking in Boiling Water Reactors (BWRs)

The following information pertaining to this event is also being reported concurrently in the Federal Register. Appendix A (see Example 2 of "For Commercial Nuclear Power Plants") of this report notes that major degradation of the primary coolant pressure boundary can be considered an abnormal occurrence. In addition, Example 12 of "For All Licensees" notes that incidents with implications for similar facilities (generic incidents) which create major safety concern can be considered an abnormal occurrence.

Date and Place - Beginning in March 1982, at Nine Mile Point Unit 1, major cracking in large diameter piping has occurred in several boiling water reactors (BWRs).

Nature and Probable Consequences - Cracking in austenitic stainless steel piping in BWRs has been observed for many years. However, on March 23, 1982 the Niagara Mohawk Power Corporation reported an event involving leakage from welds on two nozzles connecting recirculation system piping to the reactor vessel of Nine Mile Point Unit 1 which 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. Subsequent inspections and evaluations showed extensive intergranular stress corrosion cracking (IGSCC) in heat affected zones near weld areas of the large (28-inch) diameter reactor coolant recirculation system. The licensee decided to replace the recirculation piping in all five recirculation loops, all ten safe ends, and branch piping as warranted. The replacement material is of a type less susceptible to IGSCC. The findings at Nine Mile Point Unit 1 were the first examples of major cracking in large diameter piping in the United States (cracking in large diameter piping had been reported on some foreign reactors).

The NRC issued Inspection and Enforcement (IE) Bulletin No. 82-03, Revision 1 (Ref. 1) in October 1982 for action by nine BWR plants scheduled for refueling outages in late 1982 and early 1983. Inspections pursuant to this Bulletin showed cracking in five of the first seven plants examined, prompting issuance of IE Bulletin No. 83-02 in March 1983 (Ref. 2). This Bulletin required augmented inspection of welds in the recirculation system piping, using ultrasonic testing (UT) inspection procedures of demonstrated effectiveness, for all plants beyond those identified in Bulletin No. 82-03, Revision 1, at their next refueling or extended outage but no later than January 1984. No indications of pipe cracking were found at Quad Cities Unit 1, Millstone Unit 1, Oyster Creek, Big Rock Point, and Duane Arnold. At FitzPatrick one defect was characterized as probably due to IGSCC; however, after multiple inspections the defect was determined to be well within NRC acceptance criteria for continued operation without repair.

In conjunction with these Bulletins, joint efforts by the NRC and industry have been underway to train and qualify inspection personnel, using improved UT procedures on well-characterized pipe cracks in pipe segments removed from Nine Mile Point Unit 1, to assure higher reliability in the inspection process. Although this has considerably upgraded the reliability of UT in crack detection field situations, there still remains concern about the ability of current UT procedures, in field situations, to adequately characterize the depth of identified cracks although it is believed that the discovery of cracking, where it exists, is probable.

Inspections conducted in response to these Bulletins, and other inspections, have revealed extensive cracking both in large diameter recirculation and residual heat removal (RHR) system piping welds. In Orders issued to certain plants on August 26, 1983, as discussed below, inspections were mandated for susceptible systems for 4" diameter and larger pipes.

Table 1 is a summary of the cracking observations from BWRs where piping has been examined and defects found. The summary is as of late October 1983 and indicates the extent of cracking in large diameter recirculation and RHR system piping. For the plants listed in Table 1, the total number of welds range from about 100 to 135 per plant.

Although IGSCC in the sensitized material of the heat-affected zone in BWR piping is influenced by the environmental conditions existing in the BWR reactor coolant system and stresses in the piping, including residual stresses induced by welding, there is no clear correlation between extent of cracking and operating time. Some plants with a relatively brief operating history, e.g., Hatch Unit 2, show extensive cracking. The licensee for Hatch Unit 2, Georgia Power Company, will replace the affected piping in 1984.

The pipe cracks represent a degradation from the original condition of one of the primary boundaries for the containment of radioactive material. As discussed above, cracking in austenitic stainless steel piping in BWRs has been observed for many years. Prior to Nine Mile Point Unit 1 in March 1982, however, the cracking had not occurred in large diameter piping in United States reactors. Generally, the probable consequences of small cracks is crack propagation and minor leakage of primary coolant. When small but measurable leaks occur, leakage monitoring systems detect the change of leak rate, and a plant shutdown is required if allowable leak rate limits are exceeded. Licensees are also required to perform periodic inspections of piping to detect evidence of pipe leaks. Redundant core cooling systems are available to provide cooling of the core even in the remote case of a pipe failure.

However, the Nine Mile Point Unit 1 results and subsequent inspections performed on other BWRs resulted in increased safety concern regarding the extensive range of pipe sizes involved, the large number of plants affected, the size and number of cracks, adequacy of detection and characterization of such cracks, repair techniques, and adequacy of licensees' compensatory measures

TABLE 1
Summary of Piping Weld Crack Observations
(Data as of Late-October 1983)

Plant Name	Licensee	Plant Location	12" through 28" Pipe Welds No. Examined	No. Defective
Browns Ferry Unit 1	Tennessee Valley Authority	Limestone County, Alabama	123	47
Browns Ferry Unit 2	Tennessee Valley Authority	Limestone County, Alabama	34	2
Brunswick Unit 1	Carolina Power & Light Co.	Brunswick County, No. Carolina	32	3
Cooper	Nebraska Public Power District	Nemaha County, Nebraska	135	22
Dresden Unit 2	Commonwealth Edison Company	Grundy County, Illinois	51	10
FitzPatrick	Power Authority of the State of N.Y.	Oswego County, New York	55	1
Hatch Unit 1	Georgia Power Company	Appling County, Georgia	58	7
Hatch Unit 2	Georgia Power Company	Appling County, Georgia	108	39
Monticello	Northern States Power Company	Wright County, Minnesota	135	6
Oyster Creek	Jersey Central Power & Light Co.	Ocean County, New Jersey	31	0
Peach Bottom Unit 2	Philadelphia Electric Company	York County, Pennsylvania	123	20
Peach Bottom Unit 3	Philadelphia Electric Company	York County, Pennsylvania	111	15
Quad Cities Unit 2*	Commonwealth Edison Company	Rock Island County, Illinois	86	8
Vermont Yankee	Vermont Yankee Nuclear Power Co.	Windham County, Vermont	60	34

*Preliminary results - still being evaluated.

(leak detection capability, emergency core cooling system availability, and operator training).

Causes or Causes - As discussed previously, the cracking has been determined to be the result of intergranular stress corrosion of the piping. Investigations of the basic causes of such corrosion are being made, however they are not yet fully understood.

Actions Taken to Prevent Recurrence

Licensees/Vendors - Inspections of piping either have been or are being made in accordance with IE Bulletin Nos. 82-03 Revision 1 and 83-02. Where cracking is observed, resolution is in accordance with NRC requirements, as discussed below. Efforts are underway to train and qualify inspection personnel, using improved UT procedures, to assure higher reliability in crack detection and sizing. Electric Power Research Institute (EPRI) is involved in programs for the detection and characterization of cracks, and working with the licensees in formulating qualification programs for weld inspectors. Included in EPRI's efforts is a "round robin" program to compare crack depth measurements made by UT versus results of actual destructive examinations. The purpose of this program is not only to improve UT crack detection methodology, but to train inspectors in this methodology. The NRC is participating in this program.

General Electric, the nuclear steam supply system vendor for the BWRs, is also involved by studying field and laboratory data on cracks caused by intergranular stress corrosion, rate of crack propagation, etc.

For the licensees which had not yet made inspections required by the IE Bulletins, interim compensatory measures (e.g., improved leak detection capability, ECCS availability, operator training) were established where necessary.

NRC - The NRC is closely involved in the licensees' and the vendors' efforts to assure proper detection, characterization, and resolution of the cracking problem. The NRC staff has been reviewing the inspection results of each plant on a case-by-case basis. In general, for the plants where such cracking has been observed, repairs, analysis and/or additional surveillance conditions were required. Where repair was proposed, consideration was given to the strength (relative to ASME Code margin) of the repair, its effect on the piping system, and further inspectability. Where repair was not proposed, consideration was given to uncertainties in the measurements of cracking depth and to projected growth of cracks during subsequent operation. NRC staff evaluation criteria require maintaining the inherent factor of safety prescribed by Section III of the ASME Boiler and Pressure Vessel Code for normal and faulted conditions with consideration of the uncertainties in crack size and growth rate.

As of early July 1983, five plants (Browns Ferry Unit 3, Brunswick Unit 2, Dresden Unit 3, Pilgrim Unit 1, and Quad Cities Unit 2) had not yet begun inspections. These plants were scheduled for inspections at various times from August 1983 through January 1984. However, the NRC concluded that these uninspected facilities may have similar IGSCC, which may be unacceptable for continued safe operation without inspections and repair or replacement of the affected pipes and additional surveillance requirements. Therefore, on July 21, 1983, the NRC sent letters to the licensees of the five uninspected plants requesting that by August 4, 1983 the licensees submit information regarding justification for continued operations, costs and impact of conducting the inspections on an accelerated schedule, availability of qualified personnel, and other bases to support their previously established schedules for IGSCC inspections.

On August 4, 1983, EPRI presented to the NRC staff the results of their "round robin" UT program to compare crack depth measurements made by UT versus actual destructive examination. Also on August 4, 1983, the NRC staff met with representatives from General Electric. On August 8 and 9, 1983, the NRC staff met with licensee representatives from the five BWR plants yet to be inspected to discuss their responses to the NRC letters. As a result of the meeting with the five licensees, accelerated schedules for inspections and interim additional compensatory measures (improved leak detection capability, emergency core cooling system availability, and operator training) were committed to by the licensees. The staff evaluated the information and commitments received from the licensees. On August 24, 1983, the NRC staff met with the Commission and advised them of its intent to issue Orders for each of the five plants that would confirm these accelerated inspection schedules and impose new interim compensatory measures, or confirm compensatory measures proposed by the licensees. On August 26, 1983, Orders were issued to each of the five plants. Of these five plants, preliminary inspection results as of late October 1983 were only available for one plant, Quad Cities Unit 2; these results are shown in Table 1.

On September 14, 1983, the NRC Executive Director for Operations requested the existing NRC Piping Review Committee to expand its activities into the BWR pipe crack area. The Committee is integrating its work with that of industry. The goal of this work is to develop future inspection programs and to determine the best course of action extending from inspection to long-term resolution. On October 3, 1983, the NRC Commissioners were briefed on BWR pipe crack issues. Throughout the month of October 1983, the NRC staff drafted requirements for reinspection of plants inspected under the provisions of the IE Bulletins, and criteria for repair and/or replacement of piping. At a meeting with BWR licensees on October 21, 1983, the NRC staff described the development of these plans and brought the industry up-to-date on the pipe crack issues. At the same meeting, the licensees described their past and planned future actions regarding inspection, repair, and replacement. These meetings with licensees as a group, and individual meetings with licensees to discuss specific proposals, will continue in late October and into November 1983.

Future reports will be made as appropriate.

* * * * *

83-6 Uncontrolled Leakage of Reactor Coolant Outside Primary Containment

The following information pertaining to this event is also being reported concurrently in the Federal Register. Appendix A (see Example 2 of "For Commercial Nuclear Power Plants") notes that major degradation of fuel integrity, primary coolant pressure boundary, or primary containment boundary can be considered an abnormal occurrence.

Date and Place - During August 1983, the NRC staff completed a preliminary report (Ref. 3) of a plant systems interaction event which occurred at Edwin I. Hatch Unit 2 on August 25, 1982. As described in the report, a complex series of systems interactions which followed during post-scram recovery operations resulted in a sustained and uncontrolled loss of hot pressurized reactor coolant outside primary containment and had the potential to threaten the operation of certain vital equipment. Hatch Unit 2, a boiling water reactor nuclear power plant, is operated by Georgia Power Company and is located in Appling County, Georgia.

Nature and Probable Consequences - On August 25, 1982, during power operation, the main valve disk of the "C" main steam line isolation valve (MSIV) separated from the valve stem resulting in the valve closing unexpectedly. The closure of the MSIV caused a reactor scram from high flux due to the pressure increase associated with the shut valve, and a Group 1 isolation caused by increased steam flow in the three steam lines which remained open. The Group I isolation automatically closed all MSIVs.

With the reactor scrammed and isolated, pressure began to increase quickly towards the opening pressure of the safety relief valves (SRVs). By a combination of automatic and manual opening of two SRVs, reactor pressure was brought back down to approximately 900 psig. The reactor scram and vessel isolation also resulted in a rapid shrinkage of vessel water level down to the low-low level setpoint initiating both the high pressure coolant injection (HPCI) system and the reactor core isolation cooling (RCIC) system. However, the combination of injection flow coastdown from the turbine-driven reactor feed pumps and SRV operation quickly brought water level back up to the high level trip setpoints for HPCI and RCIC. Accordingly, even though both systems had autostarted, no injection occurred prior to tripping off-line.

With level restored and pressure stabilized, the control room operators prepared to reopen the closed MSIVs by first resetting the Group 1 isolation signal which had cleared. Isolation reset allowed pressure equalization around the closed MSIVs via the steam line drain lines which had also isolated during the event. When all initial reactor trip conditions had cleared, the operators reset the scram allowing the scram discharge volumes to begin draining and depressurizing. By this time the RCIC system was manually restarted for level control of the isolated vessel. However, inventory loss through the main steam line drain lines resulted in a low reactor water level alarm condition even though RCIC was operating. When this occurred, HPCI was manually started to restore water level.

During the scram, the scram discharge volume drain line isolation valve, which received a close signal, did not fully close. The result of this malfunction, which was caused by a loose valve body-to-operator yoke, was that an open flow path existed between the reactor coolant system and the reactor building equipment drainage system. Operating personnel observed that fluid temperature and level in the reactor building equipment drain sump were rising well beyond normal operating values. Based on the overall indications in the reactor building, operating personnel concluded that hot scram exhaust water from the still pressurized reactor was discharging at high pressure into the reactor building equipment drainage system. To terminate the discharge of high temperature fluid into the reactor building, the control room operators realized that it would be necessary to reset the scram which would close the outlet scram valve and effectively isolate the reactor coolant system from the reactor building equipment drainage system.

Normal reset of the scram was not possible, however, because shortly after the scram, drywell pressure had risen above the high pressure scram setpoint initiating a second scram signal which was still in effect. This second scram signal had occurred because, as the operators were maintaining pressure by use of safety relief valves, it is surmised that one of the safety relief valve tail pipe vacuum breakers malfunctioned and allowed a momentary steam release into the drywell which pressurized the drywell to above the drywell high pressure scram setpoint. Another complication arose in that the high drywell pressure also initiated a load shedding logic which secured electrical power to the drywell chiller units which would have been the normal means of reducing the high drywell pressure. The load shedding logic also tripped the control rod drive (CRD) pumps, resulting in a loss of cooling flow to the CRD seals. Eventually, CRD temperatures increased to over 500°F, compared to their normal operating range of 160°F to 200°F; however, there was no indication of damage to the CRD seals.

Meanwhile, the RCIC system, which was being used to maintain reactor vessel water level, malfunctioned and isolated on an erroneous high turbine exhaust diaphragm pressure signal while it was injecting into the vessel. This isolation was caused by instrument drift which occurred due to abnormally high temperatures in the RCIC equipment room. These abnormally high temperatures, in turn, were caused by the release of steam from the equippment drainage system to the RCIC room via an opening in the drainage system caused by a missing threaded stainless steel pipe cap. The cap normally was installed on a short drainage hub located in the RCIC room. The steam in the drainage system was the result of the blowdown through the partially open scram discharge volume drain valve to the drainage system.

Operations personnel started a reactor feed pump and used the feedwater system and main condenser to maintain reactor vessel level. The high drywell pressure signal was electrically jumpered and the drywell chiller unit restarted. This action reduced drywell pressure to the point where the reactor scram caused by high drywell pressure could be reset. When this action was accomplished, the leakage of the reactor coolant system to the reactor building equipment drainage system was halted. The total elapsed time from the initial reactor scram until the second scram was reset, was approximately 3½ hours.

The event is significant in that it resulted in sustained and uncontrolled leakage of the reactor coolant outside primary containment. The event indicated the potential for a serious and simultaneous degradation of both the reactor coolant pressure boundary and the primary containment boundary. Primary coolant discharged through a partially stuck-open scram discharge volume drain line isolation valve into the equipment drain system, subsequently discharging to the open areas of the reactor building through an open drain hub. The scram discharge volume drain line utilizes 2-inch diameter piping. Even though the isolation valve was only partially open, this represented a direct flow path for the primary coolant and indicates the potential for an even more significant degradation of the primary coolant boundary. The resultant harsh environment in the reactor building shut down the operating RCIC system (a system important to safety); had the valve failed completely and had the leakage been larger or significantly prolonged, the possibility existed that other vital equipment located in the reactor building could have been threatened. During the event, adequate core cooling capability was available to protect fuel integrity.

Cause or Causes - Several otherwise unrelated failures combined to cause the complex chain of events which occurred. As discussed above, a main steam line isolation valve closed unexpectedly when the main valve disk separated from the valve stem. This was caused by disengagement of the poppet from the stem. The loss of the drywell chiller units occurred when they were tripped off-line because of load shedding logic associated with their safety buses. This load shedding feature was provided to prevent a potential faulted condition associated with the nonseismically qualified and nonenvironmentally qualified chiller equipment from adversely affecting the emergency power supplies during a postulated loss of coolant accident inside containment. The safety relief valve discharge to the drywell is believed to have been caused when the valve opened normally and its associated tail pipe vacuum breaker stuck in an open or partially open position. Thus, when the valve lifted a second time, the stuck open vacuum breaker allowed steam to be released directly into the drywell. The scram discharge volume drain valve failure was caused by a loose valve body-to-operator yoke which prevented the attached air operator from seating the valve plug tightly into its seat. Finally, the missing RCIC room equipment drain hub cover was probably removed several months earlier during RCIC room equipment maintenance or testing activities. Removal of this cover allowed hot steam to emanate from the opening, which wetted down and significantly increased the temperature of the electrical equipment and devices located in the room. The increased temperature also set off the fire suppression system sprinkler head located above the drain system opening. These adverse conditions caused instrument drift of devices located in the room, including the trip setting for the Barksdale pressure switch which was used for the RCIC turbine exhaust diaphragm high pressure isolation function. This switch's setpoint was found to have drifted from 8 psig to 0 psig.

Actions Taken to Prevent Recurrence

Licensee - The main steam line isolation valve manufacturer, Rockwell International, had investigated the cause of similar, earlier, valve failures at Hatch and other facilities and had recommended three potential solutions to disk-to-stem disassembly problem for the Rockwell valves. These recommended actions had either not been finalized or not been adequately evaluated and implemented for Hatch at the time of the event. The licensee has replaced the entire disk and stem assembly in both the inboard and the outboard isolation valves on the "C" steam line. In addition, the licensee plans to implement the MSIV lockpin installation discussed in General Electric Service Information Letter #224 as recommended by the valve supplier. This work will probably be accomplished in the upcoming Unit 2 refueling outage; furthermore, a procedure will be issued requiring MSIV inspection during each refueling outage after these modifications are completed.

Regarding the scram discharge volume drain valve failure, the licensee had earlier, in February 1981, proposed plant technical specification changes which would include the scram discharge volume vent and drain valves in the facility surveillance requirements. However, the proposed surveillance requirements did not meet NRC requirements, and the licensee acknowledged that revisions to the technical specifications were necessary. However, the licensee did not submit revised technical specifications, and, therefore, the revised technical specifications were not implemented at the time of the event. The licensee has since resubmitted revised Technical Specifications on September 19, 1983 and December 14, 1983, as required by a June 24, 1983 NRC confirmatory order.

Following issuance of NRC Inspection and Enforcement Information Notice No. 83-44 (Ref. 4), the licensee performed a walkdown to determine the potential for flood propagation through equipment and floor drains. This walkdown verified that drain hub caps on the 87' elevation were capped; furthermore, the hub caps have been tack welded to drain header hubs to assure they remain in place. To prevent the recurrence of a missing drain hub cap, administrative controls over drain hub caps will be upgraded. The caps will be tack welded in place and a specific maintenance authorization will be required to break the weld to remove the caps. The maintenance procedural controls involved will also be revised to specifically address the need to replace covers following completion of the activities requiring their removal.

Prior to being returned to service, those instruments associated with RCIC circuitry that experienced contact with an adverse environment were inspected, calibrated and functionally tested. As long term corrective action for the RCIC system instrument drift problem, a previously planned analog trip system incorporating transmitters and bistables will be installed to replace the mechanical switches and trip devices used in the current instrumentation and control system.

The scram discharge header drain valve that allowed escape of coolant steam into the RCIC room was inspected, disassembled, cleaned, properly reassembled and satisfactorily tested after reinstallment. The potential for loss of coolant through the scram discharge system is a generic concern and is the subject of several new NRC requirements. These include the installation of redundant scram discharge volume vent and drain valves and technical specifications for periodic surveillance of these valves. These requirements are being implemented at the Hatch units and will be complete in the near future. Implementation of these requirements should significantly reduce the probability of a recurrence of the subject event.

Loss (by design) of drywell chillers occurred due to the high drywell pressure scram. Operators were unable to reset the chillers due to the existing scram signal. No corrective actions have been pursued for this concern since manual bypasses on Engineered Safety Features are undesirable. Site personnel have been trained on bypassing signals in general (Test Shop). It is felt that this training along with the operator training on functions of systems would allow signals to be bypassed, if needed, in this or other systems on an emergency basis.

The control rod drive (CRD) pumps were lost due to the high drywell pressure scram resulting in a long period of time without cooling which caused elevated CRD seal temperatures. To allow manual restart of the CRD pumps, override switches have been installed on Unit 2 and will be installed on Unit 1 in the near future.

Since the event, safety/relief valve (SRV) functional test procedures have been revised. These revised procedures impose new surveillance requirements that call for more frequent SRV exercise including exercise under power conditions. In addition, SRV tailpipe vacuum breakers were inspected in detail to assure proper operation. A new vacuum breaker design is currently being studied for probable installation.

The NRC proposed emergency procedures guidelines for this type of event are under consideration for addition to the BWR Emergency Procedure Guidelines being developed by the BWR owners group. The procedures are expected to be completed in 1984. Training related to the procedures will commence in 1984 and should be completed in 1985.

NRC - The main steam line isolation valve disk-to-stem disassembly problem had been the subject of NRC Inspection and Enforcement Information Notice No. 81-28 issued on September 3, 1981 (Ref. 5), based on similar, earlier events.

Regarding the scram discharge volume drain valve failure, NRC had, in July 1980, based upon similar, earlier failures, requested all operating BWR licensees to propose technical specification surveillance requirements for the existing scram discharge volume vent and drain valves. The surveillance requirements were intended to be an interim measure to assure scram discharge volume vent and drain valve operability on a continuing basis during reactor operation. The NRC determined in December 1980, that long term hardware improvements in the isolation arrangements for the scram discharge volume system would also be required. As discussed above, the NRC issued a confirmatory order on June 24, 1983 regarding the surveillance requirements. The same order confirmed the licensee's commitment to install permanent scram discharge system modifications (including redundant vent and drain valves) by December 31, 1983. These modifications were developed by the BWR Owners Subgroup.

The importance of reactor building equipment drain hub covers had been identified to licensees by NRC Inspection and Enforcement Circular No. 78-06 issued on May 25, 1978 (Ref. 6). The Circular recommended that administrative controls be reviewed to assure that separation criteria were maintained and that watertight room separation devices, such as doors and hatches, were closed as appropriate. The information in the Circular was supplemented by NRC Inspection and Enforcement Information Notice No. 83-44 which was issued on July 1, 1983 (Ref. 4).

Based on a review of the previously referenced NRC staff report (Ref. 3), and a review of actions taken to date, the NRC staff will determine whether further corrective actions are appropriate.

This incident is closed for purposes of this report.

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83-7 Improper Control Rod Manipulations

The following information pertaining to this event is also being reported concurrently in the Federal Register. Appendix A (see Example 11 of "For All

Licensees") of this report notes that serious deficiency in management or procedural controls in major areas can be considered an abnormal occurrence.

Date and Place - Events at two separate licensees, involving improper control rod insertions and other violations, demonstrated breakdowns in plant management control systems designed to control operations activities and ensure safe operation of the facilities.

The first event occurred on March 10 and 11, 1983, at Quad Cities Unit 1, a boiling water reactor nuclear power plant. The plant is operated by Commonwealth Edison Company and is located in Rock Island County, Illinois.

The second event occurred on July 14, 1983, at Edwin I. Hatch Unit 2, a boiling water reactor nuclear power plant. The plant is operated by Georgia Power Company and is located in Appling County, Georgia.

Nature and Probable Consequences

Quad Cities Unit 1

On March 10 and 11, 1983, the plant was being shut down for a scheduled maintenance outage. During the day shift on March 10, the nuclear engineer requested to have the Rod Worth Minimizer (RWM) bypassed so he could load a new shutdown control rod sequence into the RWM computer. The RWM serves as a backup to procedural controls to limit control rod reactivity worth during startup and low power operation; this helps limit the reactivity addition rate in the event of a control rod drop accident. The system blocks (prevents) rod movements if the existing control rod pattern deviates from a specific sequence which was developed by the plant nuclear engineers and loaded into the RWM computer memory. Due to lower rod worths at higher power levels, the plant's procedures do not require the RWM to be operable above 30% reactor power.

After the nuclear engineer loaded the new sequence into the RWM computer, he gave the unit operator the new shutdown control rod sequence procedure (designated QTP 1600-S3, dated March 9, 1983) and a RWM control rod sequence computer printout (the printout sequence was a rod withdrawal sequence which was the reverse of the approved rod insertion sequence). The RWM was left in the bypass condition.

Following shift change, the nuclear engineer prepared a handwritten explanatory note to the sequence procedure and gave it to the evening shift unit operator and shift engineer. Reactor shutdown was to begin during the evening shift. An extra operator, scheduled for the night shift, was called in early to assist with control rod insertion because the evening shift unit operator was performing numerous surveillance tests. The extra operator reviewed the handwritten note and the computer printout and mistakenly concluded (the unit operator agreed with the extra operator's interpretation) that the rods should be inserted in the sequence listed on the RWM computer printout. As discussed previously, this sequence was the reverse of the proper sequence given in QTP 1600-S3.

At about 8:00 p.m., the extra operator began inserting control rods. By 10:15 p.m., the extra operator had inserted 33 control rods improperly; at this time, reactor power was about 30%. Contrary to procedures requiring recirculation pump speed to be manually reduced at set intervals during control rod insertion, the pumps automatically ran back to minimum speed reducing reactor power to about 20%. Also contrary to procedures, the RWM remained in a bypass condition when power was reduced below 30%.

At 11:00 p.m., the night shift came on duty. At about 11:10 p.m., the oncoming unit operator returned the RWM to service. The RWM automatically prevented additional rod movements because of the out-of-sequence control rods, but did not display any error messages because there were so many insertion errors. After failing to clear the rod block, the unit operator (after discussion with the shift engineer) declared the RWM inoperable and it was again bypassed at 11:18 p.m. The unit operator requested the extra operator to continue rod insertions. Ten more control rods were improperly inserted, reducing power from about 20% to 9%; at this point, the reactor was manually scrammed (shut down) as part of normal shutdown procedures. On the following morning, March 11, 1983, plant management discovered that the control rods had been inserted in reverse order using the RWM computer printout.

Had the plant reached very low power levels, the improper insertion of the control rods and the bypassing of the RWM could have affected the plant's ability to withstand a rod drop accident (in which a control rod suddenly drops from the reactor core, resulting in a rapid, localized increase in power and possible damage to the surrounding fuel rods). In this case, no fuel damage occurred and General Electric, the reactor vendor, determined that safety margins were not seriously degraded. The unit was manually scrammed at 9% of power, a level well above the point where safety margins would have been significantly reduced.

The event, together with numerous other violations identified by the licensee's and the NRC's investigation, however, raised concerns regarding plant management control systems designed to control operating activities and to ensure safe, controlled shutdown of the reactor.

Hatch Unit 2

On July 14, 1983, during normal startup activities from a refueling outage, the plant was operating at about 25% power. Problems with main condenser vacuum had occurred and air ejector troubleshooting had been in progress. Condenser vacuum began to decrease and the turbine was unloaded and tripped. Control rods were inserted in an attempt to reduce reactor power to within the limit of the mechanical vacuum pump so that it could be placed in service in order to maintain vacuum above the trip set point of the reactor feed pumps. A reactor feed pump low vacuum trip would cause a loss of feedwater flow to the vessel.

To reduce power more quickly, the licensee bypassed the RWM and assigned a second licensed operator to verify control rod movement as permitted by the technical specifications. At one point, the emergency rod in position switch was used to achieve the greatest possible insertion rate.

When the operator reached groups of low worth peripheral rods in the sequence, a collective discussion among the licensed operators and the supervision in the control room resulted in a decision to scram individual rods by using the individual scram switches at the scram timing panel which was already set up for scram time testing. This was not an approved procedure and resulted in the insertion of rods in an out of sequence manner. Vacuum at the time was about ½ inch above the trip point.

While the plant operator continued inserting rods at the front panel, two other operators began to insert rods at the scram timing panel with the individual scram switches. When the front panel operator observed those rods going in, he stopped inserting and verified further insertions from the scram panel. A process computer printout indicated that several rods were not fully inserted (i.e., scram toggle switches were not held down sufficiently long). These rods were subsequently rescrammed. One rod was also found in a position which was not expected based upon the rod manipulations performed by the operators. The cause of this rod being improperly positioned is not known. The vacuum pump was placed in service and vacuum stabilized at a low level. Because the one rod was improperly positioned, the reactor was scrammed as required by procedure.

The consequences of this sequence of events was operation of the reactor outside of the accident analyses contained in the plant's Final Safety Analysis Report. In addition, a control rod configuration resulted which had not been analyzed. The RWM, which is used to minimize the effects of a rod drop accident, was bypassed; the use of a second operator to verify control rod movements was apparently ineffective as evidenced by the out-of-sequence rod position.

In addition, the rod sequence control system (RSCS) was effectively bypassed. The RSCS is a backup system to the RWM and independently imposes restrictions on control rod movements to mitigate the effects of a control rod drop accident. The plant's technical specifications require the RSCS to be operable when reactor power is below 20%. However, the use of the emergency rod in position switch and the scram switches on the scram timing panel circumvented the RSCS.

Even though no fuel damage occurred, the event and related violations identified by the NRC's investigation raised concerns regarding the application of management resources to the overview of facility operations.

Cause or Causes

Quad Cities Unit 1 and Hatch Unit 2

For both events, the cause was a weakness in the plant management control systems, as evidenced by the number of procedural violations, the number and types of personnel involved, the poor judgment exercised by the control room staff, and the insufficient guidance provided by management.

Actions Taken to Prevent Recurrence

Quad Cities Unit 1

Licensee - The following corrective actions were taken pertaining to the control rod insertion error event:

1. The Station Superintendent met with each person involved in the incident to discuss with him his understanding of the event, and to personally emphasize the scope of importance of accountability for his actions. In addition, the Station Superintendent conducted accountability meetings with all plant personnel in groups.

2. A committee was formed to implement a special program to monitor all the work activities of Control Room personnel involved in the event.

3. A new system for control rod movements and sequences was established which provides clearer instructions and a better means of documentation for rod movements. To implement this system, station procedures were revised to direct responsibilities and provide instructions.

4. The RWM procedures were revised which provide better instructions for operation, sequence loading, initializing and determining operability.

5. Training was accomplished on the aforementioned procedures for control room employees.

In addition, in terms of general control room conduct, procedures and practices were reviewed and rewritten to improve the quality of interpretation, to foster adherence to all procedures, and to enhance communication among control room personnel during shift turnovers.

NRC - The NRC Region III performed a special safety inspection on March 11 through 29, 1983, of the circumstances associated with the event. Three Severity Level III violations were identified involving failures to follow shutdown procedures, to accurately document actions completed, to record operating conditions and equipment status, to perform proper shift turnover, and to maintain proper overall perspective of facility operations.

On June 21, 1983, the NRC Region III sent a letter (Ref. 7) to the licensee enclosing a notice of violation and proposed imposition of civil penalties in the amount of $150,000. In addition, the NRC letter expressed the NRC's concern over the performance of certain operating personnel during the event. A special enforcement conference was held on October 20, 1983, between those individuals and senior NRC management to discuss the Quad Cities performance. A separate enforcement conference was held previously with the licensee's management on March 29, 1983. On August 12, 1983, the licensee paid the civil penalty and described the corrective actions taken (Ref. 8). The corrective measures will be examined during future NRC inspections.

Hatch Unit 2

Licensee - Upon being notified by the NRC Resident Inspector (as discussed further below) of individual rods being scrammed from the scram timing panel without authorized procedures, senior on-site plant management immediately relieved all involved operators and shift technical advisors of control room duties. Senior licensee management counselled the individuals on their improper actions. Appropriate procedures, simulator and other training techniques, and other orders to control room personnel either have been or will be modified to clarify corrective actions and to prohibit those actions which resulted in the event. The licensee also conducted a "lessons learned" program for operators during the week of August 4, 1983. Further actions may be necessary in response to pending NRC enforcement action.

NRC - The NRC Region II performed a special inspection on July 14 and 15, 1983 of the circumstances associated with the event. Three violations were identified involving failure to follow procedures for reactor operations, and failures pertaining to operation of the RWM and RSCS. Collectively, the violations were evaluated as a Severity Level II problem (Supplement I).

The NRC participated in the licensee's "lessons learned" program to discuss the event from the perspective of the NRC. An enforcement conference was held in early November between licensee and NRC personnel. Three sessions were conducted: the first with non-supervisory senior reactor operators, reactor operators, and shift technical advisors; the second with supervisory and non-supervisory personnel involved with the event; and the third with corporate and plant management.

On December 27, 1983, the NRC Region II sent a letter (Ref. 9) to the licensee enclosing a notice of violation and proposed imposition of civil penalties in the amount of $100,000. In addition, the NRC letter requested the licensee to address a number of questions regarding plant operations and individual responsibilities. The licensee responded on January 25, 1984, including payment of the civil penalty.

On November 3, 1983, the NRC issued Inspection and Enforcement Information Notice No. 83-75 to inform licensees of the Quad Cities Unit 1 and Hatch Unit 2 events (Ref. 10).

This incident is closed for purposes of this report.

FUEL CYCLE FACILITIES

(Other than Nuclear Power Plants)

The NRC is reviewing events reported by these licensees during the third calendar quarter of 1983. As of the date of this report, the NRC had not determined that any events were abnormal occurrences.

OTHER NRC LICENSEES

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

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

The NRC is reviewing events reported by these licensees during the third calendar quarter of 1983. As of the date of this report, the NRC had determined that the following were abnormal occurrences.

83-8 Overexposure of Radiation Workers' Hands

The following information pertaining to this event is also being reported concurrently in the Federal Register. Appendix A (see Example 11 of "For All Licensees") of this report notes that serious deficiency in management controls in major areas can be considered an abnormal occurrence. In addition, Example 12 (of "For All Licensees") notes that recurring incidents which create major safety concern can be considered an abnormal occurrence.

Date and Place - During the fourth quarter of 1982 and first quarter of 1983, several foundry workers employed by Nuclear Metals, Inc., of Concord, Massachusetts, received exposures to their hands estimated at 125 rems. It is possible that overexposures to their hands also occurred prior to the fourth quarter of 1982. However, this could not be determined from available data at the time the event was first reported.

Nature and Probable Consequences - Nuclear Metals, Inc., has performed essentially the same work, as described below, with depleted uranium for several years. In the past three years, the number and size of melts conducted in the foundry have increased substantially.

The licensee receives depleted uranium metal that is sent to the foundry for melting, alloying, and casting. The melting is performed in a graphite crucible in a vacuum furnace. Foundry workers load the uranium into the crucibles, place fire brick on top of the crucibles, and load them into the furnaces. After a liquid state is reached, the metal is poured from the bottom of the crucibles into castings. The foundry workers, wearing leather gloves, remove the fire bricks and crucibles from the furnace and clean them before they are reused.

The beta dose rate at the surface of uranium metal is typically 230 millirads per hour or less. However, when uranium is melted, uranium decay products (primarily thorium-234 and protactinium-234m, both beta emitters) are physically separated. When the melted uranium is poured, quantities of these decay products remain behind, coating the crucibles, fire bricks, and inside of the furnaces. The beta dose rate from these decay products is much higher than that of the original uranium. In addition, these decay products are loose and transferable, such as to the leather gloves worn by the workers while handling and cleaning the contaminated fire bricks and crucibles. The majority of the dose rate from the contamination is contributed by the protactinium-234m which emits a beta particle with a maximum energy of 2.28 MeV.

During May 1983, a licensee representative notified the NRC Region I that they had discovered a problem involving hand contamination of workers in the foundry. The problem was described as recently identified and involved inability to decontaminate workers' hands. The representative also stated that recent measurements indicated higher radiation doses to workers' hands than had previously been measured.

The NRC Region I conducted inspections on May 26-27 and June 8-10, 1983 to review these matters. The inspectors determined from interviews with members of the licensee's health physics staff and foundry workers that in November 1982 the health physics staff identified that the leather gloves worn by the foundry workers were routinely contaminated with uranium decay products which produced high beta dose rates inside the gloves. Licensee representatives stated that one reason why contamination levels and resulting radiation levels might have been higher during this time period than previously was the implementation of a policy allowing only three pairs of leather gloves per day per worker. While foundry workers were provided with wrist badges during 1982 and the first quarter of 1983, these badges did not adequately measure the exposure to the workers' hands.

Additional evaluation of the exposure to the workers' hands were not made until March of 1983. In March 1983, dosimeters were placed on the hands of foundry workers and four workers were removed from work in the foundry because their measured exposure exceeded the licensee's administrative limit of 12.5 rem during the first quarter of 1983.

On June 9, 1983, NRC inspectors obtained a contaminated glove and made measurements of the dose rate on the inside of the glove to assist in determining the probable exposure to the hands of foundry workers during the fourth quarter of 1982 and first quarter of 1983. The licensee made identical measurements and reported the results to the NRC. The licensee agreed to transfer the contaminated glove to the Department of Energy, Idaho National Engineering Laboratory (INEL) for a more precise determination of the dose rate and an identification of the radionuclides on the contaminated glove. INEL provides such analysis under contract to NRC.

Based on the INEL determination and the other measurements, the inspectors concluded that the inside surface dose rate on a typical glove was approximately 960 millirems per hour. Interviews with foundry workers indicated that they typically wore such gloves for 10 hours per week. The inspectors concluded that the typical extremity dose was 9.6 rems per week or 125 rems per quarter for 10-15 foundry personnel. NRC regulations limit the dose to the extremities to not more than 18.75 rems per calendar quarter. In March 1983 the licensee required the use of better extremity dosimetry, the simultaneous use of multiple gloves and other engineering controls.

Based on further evaluations performed by the licensee, and submitted to the NRC on October 14, 1983, the licensee concluded that 16 workers received between 19.8 and 143 rems to the hands during both the fourth quarter of 1982 and the first quarter of 1983. These estimates are in relative agreement with the NRC estimates, considering the potential errors involved. The licensee further estimated that the workers each received between 1000 rems and 2200 rems to the hands over the past six years.

The NRC medical consultant reports that no visible damage has occurred to the worker's hands; however, he will continue to review the case.

Cause or Causes - Weaknesses in the management control of the licensee's radiation safety program resulted in inadequate evaluation of the exposures to the workers' hands and assignment of inadequate extremity dosimetry. In addition, implementation of the policy allowing only three pairs of gloves per worker per day may have produced higher contamination levels and resulting higher radiation levels on the gloves than normal. The exposures received could have been considerably reduced had timely management actions been taken after the problem was first identified.

Actions Taken to Prevent Recurrence

Licensee - The licensee has assigned hand dosimetry (ring thermoluminescent dosimeters) to each individual, provided additional protective clothing, required frequent changing of contaminated gloves, provided remote handling tools and implemented engineering controls. The health physics technician assigned to the area is monitoring work closely and the health physics staff is monitoring measured exposures to assure no exposures in excess of the limits occur.

On July 22, 1983, the licensee submitted a preliminary report to the NRC Region I regarding an evaluation of the exposures received by the workers. A more complete evaluation was submitted on October 14, 1983.

NRC - An enforcement conference was held with the licensee at the Region I office on July 27, 1983. A follow-up management meeting was held at the licensee's facility on August 2, 1983. A letter confirming the licensee's planned actions to strengthen their radiation safety program was sent on August 5, 1983.

On September 1, 1983, the NRC sent the licensee a Notice of Violation and Proposed Imposition of Civil Penalty in the amount of $9,600 (Ref. 11). Several violations were identified, including radiation exposures in excess of regulatory limits to the skin of the hands of the workers. On September 30, 1983, the licensee forwarded a letter describing corrective actions. These corrective actions, and their effectiveness, will be examined by the NRC during subsequent inspections. In addition, the licensee paid the civil penalty.

NRC Inspection and Enforcement Information Notice No. 83-73 (Ref. 12) was issued on October 31, 1983 to inform appropriate licensees of the event. Suggestions were made to the licensees to help prevent similar problems.

This incident is closed for purposes of this report.

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83-9 Willful Violation of License and Material False Statement to the NRC

The following information pertaining to this event is also being reported concurrently in the Federal Register. Appendix A (see Example 11 of "For All Licensees") notes that serious deficiency in management or procedural controls in major areas can be considered an abnormal occurrence.

Date and Place - On January 17, 1983, during a routine inspection of American Testing Laboratories, Inc., in Salt Lake City, Utah, licensee management made a material false statement regarding use of licensed material. Further, during a subsequent investigation, it was found that the licensee had willfully violated certain license conditions.

Nature and Probable Consequences - During the NRC inspection on January 17, 1983, the licensee's laboratory manager stated to the inspector that all licensed material had been in storage and had not been used. The inspector, therefore, did not review licensee activities and records in regard to license conditions governing the use of portable moisture/density gauges and an asphalt content gauge. The gauges contained sealed radioactive sources not exceeding 10 millicuries of cesium-137 and 330 millicuries of americium-241.

Following the inspection, the NRC Region IV office received allegations that, at the time of the inspection, the licensee was using three gauges. As a consequence of these allegations, an investigation of the licensee's facility at Salt Lake City, Utah, was conducted May 23-25, 1983, by representatives of the NRC Office of Investigations Field Office in Region IV. The results of this investigation indicated that at the time of the January inspection, one of the gauges was in use and, in fact, from the time the NRC license was issued, the gauges had been used repeatedly in conducting licensed activities. The licensee's laboratory manager admitted in a sworn statement that licensed material had been in use at the time of the previous inspection. Three violations of NRC radiation safety regulations and license conditions were also identified during the inspection including: (1) failure to perform sealed source leak tests at proper intervals, (2) failure to institute an external dosimetry program, and (3) failure to use an approved shipping container and to block and brace the container used during transport.

Cause or Causes - As previously stated, licensee management had willfully violated certain license conditions ever since the license was issued. In addition, licensee management made a material false statement to the NRC regarding use of licensed material.

Actions Taken to Prevent Recurrence

Licensee - The licensee responded to the NRC Order, described below, by letter dated June 23, 1983, wherein was made a commitment to honesty during future dealings with the NRC and a commitment to implement corrective actions for the safety-related violations identified during the NRC investigation.

NRC - As a result of the NRC investigation, an Order to Show Cause and Order Temporarily Suspending License (Effective Immediately) was issued to the licensee on June 10, 1983 (Ref. 13). An enforcement conference was held with licensee management at the NRC Region IV Office on June 14, 1983. The licensee responded to the Order to Show Cause on June 23, 1983. The licensee responded to each of the items of noncompliance cited in the Order and described corrective actions planned to preclude recurrence of the violations. An inspection of the licensee's premises on July 26, 1983, confirmed that licensed material had been secured and apparently had been stored in compliance with the Order Temporarily Suspending License.

The NRC examined the licensee's response and concluded that the license should be revoked. An Order Revoking License was sent to the licensee on December 16, 1983 (Ref. 14).

This incident is closed for purposes of this report.

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83-10 Overexposure to a Radiographer's Hand

Preliminary information pertaining to this event was reported in the Federal Register (Ref. 15). Appendix A (see Example 1 of "For All Licensees") of this report notes that exposure of the feet, ankles, hands, or forearms of any individual to 375 rems or more of radiation can be considered an abnormal occurrence.

Date and Place - On June 15, 1983, NRC Region I was notified by Automation Industries, Inc., of Phoenixville, Pennsylvania, that a ring dosimeter worn by one of its radiographers showed an exposure which exceeded the extremity dose limit of 18.75 rems for any calendar quarter as specified by NRC regulations in 10 CFR 20.101. The NRC estimates that the exposure was 650-1100 rems to the index finger and thumb of one hand. At the time of exposure, the Automation Industries' radiographer was performing consulting services for U.S. Testing Company, Inc., of Reading, Pennsylvania, at a temporary field site in Hoboken, New Jersey.

Nature and Probable Consequences - A radiography crew, employed by U.S. Testing Company, arrived at the work site at approximately 4:00 p.m. on June 9, 1983. The radiographers set up equipment and darkroom as necessary for the work assignment. The area was posted, barriers were established, surveys were conducted, and other pre-radiography procedures were followed. While cranking the source from the radiographic exposure device to the unshielded position, the source apparently disconnected from the drive cable and jammed in the guide tube which prevented the radiographer from retracting the source to a shielded position. The radiographers attempted to dislodge the source and move it toward the camera end of the guide tube by elevating and shaking the guide tube with the assistance of a makeshift, remote handling device fabricated from a pair of pliers attached to broomstick handles.

During the source retrieval attempt, pocket dosimeters were checked frequently, dose readings recorded, and dosimeters rezeroed prior to entry into the restricted area. When the radiographers' pocket dosimeter readings totaled approximately 450 millirems, the radiographers discontinued their attempts to retrieve the source, reported the incident to licensee management, and secured the area until the licensee's consultant (a radiographer, employed by Automation Industries) could arrive onsite to perform the source retrieval.

Upon arrival, the consultant reviewed the events that had transpired and was told by the radiographers that the source was located in the guide tube approximately two feet from the radiographic exposure device; the radiographers were unable to verify this, however, since their survey meter had gone off scale. The consultant did not conduct his own surveys to verify this information or determine independently the position of the source. Available survey instruments were not capable of recording radiation levels in excess of 1R/hr. Upon disconnecting the guide tube from the device, the consultant discovered that the source was partially lodged in the camera with only the source capsule extending from the exit portal. Remote tongs were used to retrieve the source from the exposure device to transfer the source to a source changer.

The consultant's personnel dosimetry consisted of one pocket dosimeter with a range of 0-200 millirem; another pocket dosimeter with a range of 0-1,000 millirem; a digital read-out, alarming dosimeter; a whole body dosimeter; and a ring dosimeter for each hand. The total whole body exposure reported by the digital dosimeter for the source retrieval was 185 millirems.

At the time, the consultant estimated he had received a hand exposure of 8-9 rems, and a whole body dose of about 185 millirems. When the consultant returned to his company and had his ring dosimeters processed, however, the doses indicated by these dosimeters labeled for the left and right hands were about 59 rems and 12 rems, respectively. However, it cannot be determined which hand actually received the higher exposure since the consultant could not verify that he had worn the ring dosimeters on the hands for which the dosimeters were labeled; also, he could not recall which hand he had used to disconnect the guide tube. The consultant's whole body film badge indicated 185 millirems, the same as indicated by the digital dosimeter described above.

U.S. Testing Company evaluated the extremity exposures but failed to realize that the 59 rem dose indicated by one of the ring dosimeters would not accurately reflect the actual dose received. The consultant's ring dosimeters were worn on the third finger of each hand; however, he had contacted the guide tube with his thumb and index finger. Since the radiation level falls off sharply from the distance to the source, the dose indicated by the ring dosimeter would be several orders of magnitude less than the actual dose received at the points of contact with the guide tube.

NRC evaluation of the maximum exposure to the consultant's hand indicated that his thumb and index finger received an estimated 650-1100 rems. The NRC calculations were based upon previous thermoluminescent dosimeter measurements of the gamma and secondary electron dose rates from an iridium-192 source in an identical source guide tube. A reenactment of the incident provided an estimate of the time period required to disconnect the source guide tube from the radiographic exposure device. The ring dosimeter readings actually reported are in agreement with NRC calculations if the differences in distance from the third finger (where the ring dosimeter was worn) to the edge of the index finger and thumb in contact with the guide tube are considered. It is estimated that the other hand received 12 rems as was indicated by the ring dosimeter.

The consultant's hands have been examined by a physician experienced in treatment of radiation injuries. No visible effects were observed or expected considering the estimated dose range. A blood sample was taken and showed no abnormalities. The physician does not expect any long term health effects. An NRC medical consultant has reviewed the case and agrees.

Cause or Causes - The direct cause of the overexposure was the failure to perform an adequate radiation survey to determine the actual location of the source prior to the attempt to recover it.

The cause of the source disconnect is under investigation by Region I. After the source was secured by the consultant in the source changer, the radiographic exposure device, guide tube, drive cable, and pigtail end of the source were examined by the consultant and representatives of U.S. Testing Company for defects. No defects or abnormalities were visually identified. The consultant connected a dummy source to the drive camera to check the functional operation of the radiographic exposure device system and found no functional abnormalities.

Actions Taken to Prevent Recurrence

Licensee (U.S. Testing Company) - Emergency procedures have been expanded to specifically include a description of emergency procedures for source disconnects. The radiographers involved in this particular incident have been instructed in appropriate actions that should have been taken. Management agreed that this particular incident would be written up and distributed to all radiographers during upcoming training or refresher training sessions for radiographers of all levels of qualification throughout the company.

NRC - The NRC conducted an investigation on June 22 and 23, 1983, to review the circumstances associated with the event. The NRC performed calculations to better characterize the actual exposure received by the consultant's hands. An NRC medical consultant was requested to review the possible health effects of the overexposure. The investigation of the reasons for the source disconnect is continuing.

The NRC inspection report was sent to U.S. Testing Company on July 29, 1983. Five violations were noted: overexposure of an individual's hand; failure to perform an accurate radiation survey; failure to adequately evaluate the actual exposure received in the source recovery; failure to adequately train an individual who performed a source recovery; and failure to follow required emergency procedures. U.S. Testing Company is responsible for the violations since Automation Industries, Inc. was acting as their consultant. Automation Industries is not licensed to perform field work.

An enforcement conference was held with representatives of U.S. Testing Company at the Region I office on August 3, 1983, to discuss the violations and the licensee's proposed corrective actions. On October 7, 1983, the NRC issued a Notice of Violation and Proposed Imposition of Civil Penalty in the amount of eight thousand dollars. The licensee responded to the Notice in a letter dated October 27, 1983. After careful consideration of the licensee's response, the NRC staff concluded that the violations did occur as described in the Notice. Accordingly, an Order Imposing Civil Monetary Penalty in the amount of $8,000 was issued to the licensee on January 10, 1984, which the licensee subsequently paid. In addition, preparation of an Inspection and Enforcement Notice to inform all licensees performing radiography of this event is under consideration.

This incident is closed for purposes of this report.

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83-11 Radiation Overexposure

The following information pertaining to this event is also being reported concurrently in the Federal Register. Appendix A (see Example 1 of "For All Licensees") of this report notes that exposure of the whole body of any individual to 25 rems or more of radiation can be considered an abnormal occurrence.

Date and Place - On July 29, 1983, Kay-Ray, Inc., Arlington Heights, Illinois, an industrial gauge manufacturer and distributor, reported that one of its employees had received whole body and hand radiation overexposures. The licensee had previously reported on May 24, 1983, that another employee had received an overexposure to his hands.

Nature and Probable Consequences - The July overexposure involved an employee whose duties included loading sealed radiation sources into industrial gauges. The film badges worn during the period July 18-24, 1983, and subsequent evaluation by NRC Region III inspectors indicated a whole body radiation exposure of 25.3 rems (14.4 rems gamma radiation and 10.9 rems beta radiation). The exposure to the employee's hands during the same time period was indicated to be 60.7 rems. (NRC regulations limit radiation exposure in a calendar quarter to 3 rems whole body and to 18.75 rems to the hands. A rem is a standard measure of radiation exposure.)

An NRC inspection was unable to determine the specific cause of the overexposures. No known incidents occurred during the source handling activities which would account for the radiation exposure levels. The inspectors did note that the number of source handling operations was greater than normal and that several problems were encountered by the individual in loading sources. The additional workload and the problems encountered, however, were not considered sufficient to explain the overexposures.

The employee was examined at a local hospital and blood tests were performed. There was no evidence of any radiation damage. Radiation exposures of this magnitude would not be expected to result in any medically observable effects.

The second radiation overexposure occurred in May 1983 with an employee receiving a quarterly exposure to his hand of 29.9 rems, as measured by a ring thermoluminescent detector (TLD), which measures radiation exposure.

The specific cause of the overexposure could not be determined in an NRC inspection. A contributing factor, however, may have been the use of a new procedure for removing sources from their holders in preparation for disposal. The new procedure proved to be more time-consuming and arduous than the one previously used, and the procedure has subsequently been discontinued.

Cause or Causes - While no specific incident or direct cause of these two overexposures could be determined, the overexposures and other violations identified in recent NRC inspections indicated serious weaknesses in the company's radiation protection program and its ability to ensure the safe handling of radioactive materials.

Actions Taken to Prevent Recurrence

Licensee - In response to an NRC Order issued as a result of the overexposures, the licensee has upgraded its radiation protection and management program. Source handling procedures have been revised and employees have received extensive retraining. In addition, the licensee has developed a program to audit employee performance during source loading and other activities involving radioactive materials. It has also retained a radiation protection consultant to assist it in training and other radiation protection activities.

NRC - On August 15, 1983, the NRC issued an Order suspending the NRC license of Kay-Ray, Inc., as a result of the overexposures and other violations. In addition, on September 23, 1983, a $1,800 fine was proposed for the violations, which was subsequently paid by the licensee. The suspension order was rescinded on September 16, 1983, after the licensee had submitted its plans for upgrading its radiation protection program.

An NRC inspection on October 18, 1983, determined that the upgraded radiation protection program had been satisfactorily implemented.

This incident is closed for purposes of this report.

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83-12 Diagnostic Misadministration of a Radiopharmaceutical

The following information pertaining to this event is also being reported concurrently in the Federal Register. Appendix A (see the general criterion) of this report notes that a moderate or more severe impact on the public health or safety can be considered an abnormal occurrence.

Date and Place - On August 24, 1983, the NRC Region I office was notified by Thomas Jefferson University, Philadelphia, Pennsylvania, that a patient had been orally administered 100 millicuries of technetium-99m DTPA (diethylenetriaminepentacetic acid) for the purpose of evaluating gastric emptying. The dose prescribed for this procedure was 100 microcuries of technetium-99h DTPA, which is 1000 times less than the dose actually administered.

Nature and Probable Consequences - On August 24, 1983, a patient was presented at the licensee's Nuclear Medicine Department in preparation for a gastric emptying analysis. The patient arrived prior to the Department's normal work hours. The study had been requested by the patient's attending physician in a written request which had been received in the Department on the previous day. The written request had not been reviewed by the Nuclear Medicine physician, a standard, but not required, procedure, since the Nuclear Medicine physician had not yet arrived in the Department.

Normally, it is during this review that the Nuclear Medicine physician prescribes the appropriate radiopharmaceutical and dose the patient is to receive. It is an accepted practice at this institution to proceed with a Nuclear Medicine study, without the physician review, when the requested study is a routine procedure for which a standard dose is prescribed in the Department procedure manual. A copy of the portion of the manual specifying the doses is on file in the radiopharmacy for review by the radiopharmacist when the written request has not received a physician review.

On August 24, 1983, both the radiopharmacist and the Nuclear Medicine technologist who routinely perform this procedure, were on leave from the Department. The substitute radiopharmacist, though familiar with the preparation of technetium-99m DTPA in bulk, had not prepared the radiopharmaceutical in the dose required for the gastric emptying analysis. The Nuclear Medicine technologist, who administered the dose and performed the imaging procedure, had participated in this study on approximately four other occasions. This study has been performed an average of 20 times per year for the last four years. The substitute radiopharmacist, upon referring to the dose chart in the radiopharmacy, noted that the dose was not listed on the chart. The Nuclear Medicine technologist referred the pharmacist to the Department procedure manual. The procedure in the manual contained a typographical error; the dose was written as "100 MCI of 99m Tc DTPA", meaning 100 millicuries of technetium-99m DTPA. The procedure should have read, "100uCi of 99m Tc DTPA", meaning 100 microcuries of technetium-99m DTPA. The pharmacist, though not familiar with the dose range, did question the dose listed, as it was 4 to 5 times higher than any other diagnostic radiopharmaceutical dose listed on the radiopharmacy chart. The Nuclear Medicine technologist requested that the pharmacist again review the written procedure. The Nuclear Medicine technologist did not give the pharmacist her full attention on the matter, as she was engaged in setting up the imaging equipment because of the patient's early arrival to the department. The pharmacist prepared the 100 millicurie dose of technetium-99m DTPA, which was orally administered to the patient by the technologist. Only after the imaging equipment's overresponse to the high radioactive content in the patient did the technologist realize that a misadministration had occurred.

Since less than 1% of the technetium-99m DTPA is absorbed from the digestive tract, the licensee attempted to reduce the radiation dose through emetics and laxatives. This proved ineffective since the patient had a gastric neuropathy which was not responsive to these treatments. Initial dose estimates of 200 rems to the lining of the stomach and intestinal tract were revised downward to less than 50 rems based on a more thorough evaluation of information available from the Medical Internal Radiation Dose (MIRD) calculations published by the Society of Nuclear Medicine. An NRC medical consultant concurred in these dose estimates. The patient exhibited no ill effects due to the misadministration of the radiopharmaceutical.

Cause or Causes - The direct cause of this incident was the typographical error contained in written procedure combined with the substitute radiopharmacist's unfamiliarity with the dose range associated with this procedure. A contributing cause was the patient's early arrival in the Department which altered the daily routines of both the pharmacist and the technologist in bypassing the Nuclear Medicine physician's review.

Actions Taken to Prevent Recurrence

Licensee - The licensee has eliminated all abbreviations in radiopharmaceutical dose prescriptions contained in the procedure manual and dose charts. In addition, all procedures and doses will be periodically reviewed to ensure that all information is correct and current. The licensee has also taken steps through additional training to ensure that substitute staff members are knowledgeable in both routine and special procedures when regular staff members are unavailable.

NRC - An inspection was performed to verify the licensee's corrective actions. An NRC medical consultant was retained. The consultant concurred in the estimated dose received by the patient.

This incident is closed for purposes of this report.

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83-13 Widespread Radiological Contamination

The following information pertaining to this event is also being reported concurrently in the Federal Register. Appendix A (see general criterion 1) of this report notes that moderate release of radioactive material licensed by or otherwise regulated by the Commission can be considered an abnormal occurrence. The importance of the event was enhanced by the widespread nature of the radiological contamination (including unrestricted areas) and the significant clean up efforts required.

Date and Place - On September 13, 1983, a sealed radiation source containing cesium-137 was damaged at the Shelwell Services, Inc., facility in Hebron, Ohio. The cesium contamination was spread about the Shelwell facility and subsequently carried to employees' homes and other locations in the Hebron area.

Nature and Probable Consequences - Shelwell Services, Inc., is licensed by the Nuclear Regulatory Commission for use of radiation sources in well logging activities. Well logging for gas and oil wells involves lowering a radiation source into the drilled hole and measuring the radiation reflected on the rock strata.

On September 13, 1983, three Shelwell employees were attempting to remove a sealed source containing 2 curies of cesium-137 from a source holder. The source was a stainless steel capsule about 0.34 inch in diameter and 0.75 inch long. After several attempts to free the capsule, the workers placed the cylindrical source holder on a lathe, began turning the lathe, and used a hand-held drill bit to penetrate the end of the holder. The drill bit cut into the capsule itself, allowing a portion of the contents, in the form of cesium chloride powder, to spill out.

The cesium was spread throughout the room as airborne contamination and on the shoes and clothing of the workers. Shelwell personnel attempted to clean up the contamination, but did not have adequate survey equipment, nor did they have the expertise to perform an adequate radiation survey and decontamination. The workers also failed to realize that they were carrying the cesium powder on their shoes and clothing. As a result the workers' cars and homes as well as other locations they visited were contaminated by the cesium carried on their shoes and clothing.

The licensee reported the source damage incident to the NRC's Region III Office on September 14, 1983. A Region III inspector was dispatched to the Shelwell facility and when he arrived on September 15, 1983, he determined that there was extensive cesium contamination throughout the Shelwell facility and a strong likelihood that the contamination had been spread offsite.

An additional team of four NRC inspectors and a Department of Energy representative was sent to the Shelwell site by charter aircraft on September 15 and they were joined by additional personnel from the Ohio Disaster Services Agency and the U.S. Department of Energy's Radiological Assistance Team. Preliminary surveys that night indicated that the homes of the three workers involved in the source damage accident were contaminated with the cesium powder.

Surveys by the state and federal teams determined that the contamination levels did not represent an immediate health and safety problem, but were such that the contamination should be cleaned up as a precaution.

Radioactive contamination in the three homes, and a fourth home visited by one of the workers, involved generalized contamination levels ranging up to 250 microrems per hour with spotty contamination measuring 10 to 20 millirems per hour. The highest measurement in the homes was a single isolated spot surveyed at 100 millirems per hour. (A rem is a standard measure of radiation exposure. A millirem is 1/1000th of a rem and a microrem is 1/1,000,000 of a rem. Natural background radiation typically measures 10 microrems per hour, while the NRC's limit for radiation exposure to members of the public is 2 millirems per hour.)

The state and federal survey teams later identified a total of 15 homes with cesium contamination levels which required decontamination--the four homes with the highest amounts of contamination plus ten additional homes with lesser levels. The licensee retained a radiation services contractor to decontaminate the homes, and decontamination was completed on November 14, 1983. Follow-up surveys were performed by NRC Region III to assure that the homes had been adequately decontaminated.

In addition, the survey teams checked 16 area businesses with detectable radioactivity being identified at 6 of them. This contamination involved only small areas and was readily cleaned up by the survey teams. Five individuals who had visited the Shelwell site and their vehicles were also surveyed. Minor contamination requiring cleanup was found in one vehicle.

On September 20, 1983, the NRC's Office of Inspection and Enforcement issued an Order suspending all licensed activities using radiation materials at the Shelwell site and field locations (Ref. 16). The licensee was also ordered to show cause why its license should not be revoked because of the mishandling of the cesium source and subsequent spread of contamination. The licensee was also directed to submit, for NRC approval, a plan for decontamination of its facility.

The licensee's contractor, in preparation for formulating a decontamination plan, surveyed the buildings on the Shelwell site. Building 1, a garage containing maintenance vehicles and equipment, had several isolated spots measuring 1 to 2 millirems per hour. Building 2, a storage facility where the September 13 incident took place, showed multiple areas of contamination with surface readings from 2 to 10 millirems per hour. A vacuum cleaner, apparently used by the employees to clean up the contamination after the source was damaged had a measurement of 600 millirems per hour, the highest found in the Shelwell facility.

The NRC retained a medical consultant to examine the individuals involved in the source damage incident and in the subsequent attempted cleanup activities. The three individuals who were present when the source was damaged and two additional employees who performed cleanup activities were examined at the

University of Cincinnati and checked in a whole body radiation counter. All five individuals showed some evidence of uptake (inhalation) of the cesium powder, but the levels observed were well within NRC regulatory limits of occupational exposures.

The film badges worn by the three employees involved in the source damage incident showed radiation exposures of 13 rems, 2.7 rems, and 110 millirems with the highest reading for the worker who actually handled the source in its storage tube and performed the machining work on the lathe. While two of the exposures are above the NRC occupational exposure limit of 1.25 rems per calendar quarter, they are below the point where any observable medical effects would be expected.

Cause or Causes - The damage to the source and subsequent spread of contamination was caused by inadequate source handling procedures and a lack of understanding of the hazards of radiation and contamination. Had adequate technical assistance been sought promptly, the contamination would have been limited to only a portion of the licensee's facility.

Actions Taken to Prevent Recurrence

Licensee - As described in the licensee's October 17, 1983, response to the NRC Order, all licensed radioactive material was placed in storage. Offsite decontamination was accomplished and was verified by NRC site officials on November 16, 1983, to be in compliance with NRC criteria. The licensee described a revised radiation protection program which would aid in complying with the terms of its license. In addition, the licensee described its proposed onsite decontamination plan.

NRC - Because the damage to the source and subsequent mishandling of the initial decontamination by the licensee, the NRC issued an Order on September 20, 1983 (Ref. 16), immediately suspending the license of Shelwell Services, Inc., and requiring the company to show cause why the license should not be reviewed to determine whether or not license revocation is the appropriate regulatory action.

The NRC and other state and federal agencies took prompt and effective action to minimize the offsite consequences of the spread of contamination. After approving the licensee's proposed onsite decontamination plan on October 25, 1983, the NRC has closely monitored the licensee's activities and those of its contractor in decontaminating the company's facility.

The NRC staff met with licensee representatives on October 28, 1983 to obtain additional information regarding corrective actions. Subsequently, on November 7, 1983, the NRC issued a rescission of the license suspension and modified the license to include additional conditions (Ref. 17).

The NRC issued Inspection and Enforcement Information Notice No. 83-74 on November 3, 1983 (Ref. 18), to inform NRC well logging licensees of the circumstances of the Shelwell source damage incident and subsequent contamination.

This incident is closed for purposes of this report.

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83-14 Exposure of Patients to Significantly Less Than Prescribed Therapeutic Doses

The following information pertaining to this event is also being reported concurrently in the Federal Register. Appendix A (see Example 12 of "For All Licensees") of this report notes that recurring incidents which create major safety concern can be considered an abnormal occurrence.

Date and Place - On September 27, 1983, the NRC Region I Office was notified by the University of Pittsburgh, Pittsburgh, Pennsylvania, of the undertreatment of a number of patients who had had teletherapy with cobalt-60. The undertreatments, to a total dose more than 10% lower than prescribed in the treatment plan, had occurred between July 8, 1980 and August 30, 1983.

The licensee's 30-day report dated October 28, 1983, identified a total of 53 patients who had been undertreated with total doses more than 10% below that prescribed in the treatment plan.

Nature and Probable Consequences - On August 30, 1983, the licensee recalibrated the radiation attenuation factors for all wedges used in treatment of head and neck tumors by cobalt-60 teletherapy. The individual performing the measurements reported to the Director of Physics that there was a discrepancy between the July 7, 1980 wedge factors and the August 30, 1983 wedge factors. For example, for the 60 degree wedge, the radiation attenuation factor was found to be 2.71 rather than 1.63 as had been used in treatment planning since July 8, 1980.

The Director of Physics verified the August 30, 1983 wedge factors by independent measurements on three separate occasions. Additionally, on September 21, 1983, a representative from the Radiological Physics Center, Houston, Texas, verified the wedge factors as measured on August 30, 1983. The licensee began reviewing treatment plans on August 30, 1983 to determine the effect of this error on the radiation dose delivered to tumors. As of October 3, 1983, 53 instances of delivered doses more than 10% below the prescribed dose had been identified by the licensee. Most of the 53 cases identified were 10 to 15% below the prescribed dose, however, 4 cases were 30% below the prescribed dose; over 800 treatment plans were reviewed by the licensee.

The licensee states in the October 28, 1983 report that, although patients received doses less than prescribed by the physician based on the licensee's treatment protocol, only one patient received a dose that was less than the lower limit of the dosage range normally accepted within the community of physicians specializing in radiation therapy.

A review of patient records indicated that in 47 of the 53 cases identified, there had been no further evidence of disease. The licensee stated that for the types of cancer treated, 6 out of 53, or (11%), is within the expected range of recurrence. Additionally, the licensee noted that in one of the six cases with evidence of recurrence of disease, there was concurrent diagnosis of primary carcinoma of the lung at the time head and neck carcinomas were being treated.

As of December 5, 1983, the licensee had notified all referring physicians of the patients involved.

Cause or Causes - The direct cause of this incident was either a mistake in measurement or a misrecording of a correct measurement. In addition, only a single measurement of radiation transmitted through the wedge was recorded.

Actions Taken to Prevent Recurrence - The licensee, as of August 30, 1983, corrected the radiation attenuation factor for all wedges. In addition, future calibrations of wedges will require that three measurements be made of the transmitted radiation through a wedge. The wedge will then be rotated 180 degrees and three more measurements will be made. Any discrepancies between the two sets of measurements will cause the placement of the wedge in the radiation beam to be examined and the measurements to be repeated. A second set of measurements by a different individual will be performed to verify the initial measurements. Both sets of data will be recorded in a log book for review by the Director of Physics.

The NRC performed an inspection on October 3, 1983. The NRC reviewed the licensee's evaluations in the October 28 and December 3, 1983, reports and verified that corrective actions have been taken.

This incident is closed for purposes of this report.

AGREEMENT STATE LICENSES

Procedures have been developed for the Agreement States to screen unscheduled incidents or events using the same criteria as the NRC (see Appendix A) and report the events to the NRC for inclusion in this report. During the third calendar quarter of 1983, the Agreement States reported no abnormal occurrences to the NRC.

REFERENCES

1. U.S. Nuclear Regulatory Commission, Inspection and Enforcement Bulletin No. 82-03, Revision 1, "Stress Corrosion Cracking in Thick-Wall, Large-Diameter, Stainless Steel, Recirculation System Piping at BWR Plants," October 28, 1982.*

2. U.S. Nuclear Regulatory Commission, Inspection and Enforcement Bulletin No. 83-02, "Stress Corrosion Cracking in Large-Diameter Stainless Steel Recirculation System Piping at BWR Plants," March 4, 1983.*

3. U.S. Nuclear Regulatory Commission, "Preliminary Case Study Report for the Edwin I. Hatch Unit No. 2 Plant Systems Interaction Event on August 25, 1982," by S.D. Rubin, NRC Office for Analysis and Evaluation of Operational Data, August 1983.*

4. U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 83-44, "Potential Damage to Redundant Safety Equipment as a Result of Backflow Through the Equipment and Floor Drain System," July 1, 1983.*

5. U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 81-28, "Failure of Rockwell-Edward Main Steam Isolation Valves," September 3, 1981.*

6. U.S. Nuclear Regulatory Commission, Inspection and Enforcement Circular No. 78-06, "Potential Common Mode Flooding of ECCS Equipment Rooms at BWR Facilities," May 25, 1978.*

7. Letter from James G. Keppler, Regional Administrator, NRC Region III, to James J. O'Connor, President, Commonwealth Edison Company, transmitting a Notice of Violation and Proposed Imposition of Civil Penalties, Docket No. 50-254, June 21, 1983.*

8. Letter from B.L. Thomas, Executive Vice President, Commonwealth Edison Company, to James G. Keppler, Regional Administrator, NRC Region III, Docket No. 50-254, August 12, 1983.*

9. Letter from James P. O'Reilly, Regional Administrator, NRC Region II, to R. J. Kelly, Executive Vice President, Georgia Power Company, transmitting a Notice of Violation and Proposed Imposition of Civil Penalties, Docket No. 50-366, December 27, 1983.*

10. U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 83-75, "Improper Control Rod Manipulation," November 3, 1983.*

11. Letter from Thomas E. Murley, Regional Administrator, NRC Region I, to W.B. Tuffin, President, Nuclear Metals, Inc., transmitting a Notice of Violation and Proposed Imposition of Civil Penalties, Docket No. 40-00672, September 1, 1983.*

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

12. U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 83-73, "Radiation Exposure from Gloves Contaminated with Uranium Daughter Products," October 31, 1983.*

13. Letter from Richard C. DeYoung, Director, NRC Office of Inspection and Enforcement, to Roger Shepherd, American Testing Laboratories, Inc., transmitting an Order to Show Cause and Order Temporarily Suspending License (Effective Immediately), License No. 43-12757-02, June 10, 1983.*

14. Letter from Richard C. DeYoung, Director, NRC Office of Inspection and Enforcement, to Roger Shepherd, American Testing Laboratories, Inc., transmitting an Order Revoking License, License No. 43-12757-02, December 16, 1983.*

15. U.S. Nuclear Regulatory Commission, "Abnormal Occurrence: Overexposure to a Radiographer's Hand," Federal Register, Vol. 48, No. 225, November 21, 1983, 52655-52656.

16. Letter from Richard C. DeYoung, Director, NRC Office of Inspection and Enforcement, to Shelwell Services, Incorporated, transmitting an Order to Show Cause and Order Temporarily Suspending License (Effective Immediately), License No. 34-10445-01, September 20, 1983.*

17. Letter from Richard C. DeYoung, Director, NRC Office of Inspection and Enforcement, to Shelwell Services, Incorporated, transmitting a Rescission of Suspension and Order Modifying License, License No. 34-10445-01, November 7, 1983.*

18. U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 83-74, "Rupture of Cesium-137 Source Used in Well Logging Operations," November 3, 1983.*

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

APPENDIX A

ABNORMAL OCCURRENCE CRITERIA

The following criteria for this report's abnormal occurrence determinations were set forth in an NRC policy statement published in the FEDERAL REGISTER on February 24, 1977 (Vol. 43, No. 37, pages 10950-10952).

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

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

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

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

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

For All Licensees

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

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

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

4. Radiation or contamination levels in excess of design values on packages, or loss of confinement of radioactive material such as (a) a radiation dose rate of 1,000 mrem per hour three feet from the surface of a package containing the radioactive material, or (b) release of radioactive material from a package in amounts greater than regulatory limit (10 CFR § 71.36(a)).

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

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

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

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

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

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

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

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

For Commercial Nuclear Power Plants

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

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

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

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

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

For Fuel Cycle Licenses

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

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

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

APPENDIX B

UPDATE OF PREVIOUSLY REPORTED ABNORMAL OCCURRENCES

During the July through September 1983, 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. These occurrences not now considered closed will be discussed in subsequent reports in the series.

NUCLEAR POWER PLANTS

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

This abnormal occurrence was originally reported in NUREG-75/090, "Report to the Congress on Abnormal Occurrences; January-June 1975," and updated in subsequent reports in this series, i.e., NUREG-0090-1; 0090-2; 0090-3; 0090-9; Vol. 1, No. 3; Vol.2, No. 2; Vol. 2, No. 4; Vol. 3, No. 2; Vol. 3, No. 4; Vol. 5, No. 2; Vol. 5, No. 4; Vol. 6, No. 1; and Vol. 6, No. 2.

Cracking in austenitic stainless steel piping in BWRs has been observed for many years. However, the observations beginning in March 1982 at Nine Mile Point Unit 1 (see NUREG-0090, Vol. 5, No. 2) were the first examples of major cracking in large diameter piping in the United States. Subsequent reports in this series, after NUREG-0090, Vol. 5, No. 2, have described many additional plants with such cracking.

The Commission has determined that a new abnormal occurrence should be prepared, beginning with the observations at Nine Mile Point Unit 1, to report such major cracking in large diameter piping in BWRs. This is based on such factors as, (1) the extensive range of pipe sizes involved, (2) the large number of plants affected, (3) the size and number of cracks, (4) problems in detection and characterization of such cracks, and (5) the significant efforts being expended on the issue.

This new abnormal occurrence is designated as "83-5 Large Diameter Pipe Cracking in Boiling Water Reactors" and is described in this report, beginning on page 1.

Further progress regarding the cracking problem will be reported under the new abnormal occurrence 83-5. Therefore, abnormal occurrence 75-5 is being closed out for purposes of this report.

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79-3 Nuclear Accident at Three Mile Island

This abnormal occurrence was originally reported in NUREG-0090, Vol. 2, No. 1, "Report to Congress on Abnormal Occurrences: January-March 1979," and updated in subsequent reports in this series, i.e., NUREG-0090, Vol. 2, No. 2; Vol. 2, No. 3; Vol. 2, No. 4; Vol. 3, No. 1; Vol. 3, No. 2; Vol. 3, No. 3; Vol. 3, No. 4; Vol. 4, No. 1; Vol. 4, No. 2; Vol. 4, No. 3; Vol. 4, No. 4; Vol. 5, No. 1; Vol. 5, No. 2; Vol. 5, No. 3; Vol. 5, No. 4; Vol. 6, No. 1; and Vol. 6, No. 2. It is further updated as follows.

Reactor Building Entries

During the third calendar quarter of 1983, 48 entries were made into containment. There have been a total of 295 entries since the March 28, 1979 accident. Major activities included the continued refurbishment of the polar crane, the refurbishment and preparation of the "A" spent fuel pool for the eventual staging and temporary storage of core fuel debris, the radiological characterization of the area under the reactor vessel head, and initiation of further core examinations.

EPICOR-II/Submerged Demineralizer System (SDS) Processing

The EPICOR-II system did not process any water during the third quarter of 1983. The SDS processed approximately 150,700 gallons of water.

EPICOR-II/Prefilter and SDS Liner Shipments

On July 12, 1983, the final two EPICOR-II prefilter waste liners were shipped from TMI to the Idaho National Engineering Laboratory (INEL). These prefilters were the last in a group of 50 liners that have been shipped to INEL.

During the past several years, the EPICOR system has been used to process effluent from the SDS system. As of the end of this reporting period, 31 low-level EPICOR demineralizers had been generated. The first group (4) of these demineralizers was shipped to the Hanford Washington commercial burial facility on August 19, 1983. A total of 11 demineralizers was shipped this quarter.

Reactor Building Polar Crane/Contractor-Employee Allegations

As previously reported, allegations were made by a GPUNC contractor (Bechtel) employee relating to mismanagement, NRC/licensee collusion, unsafe modifications, and harassment. Subsequent to the initial allegation, two GPUNC employees also submitted affidavits relating to the same subjects.

On March 25, 1983, NRC Chairman Palladino instructed the NRC's Office of Investigations (OI) and Office of Inspector and Auditor to investigate and address the allegations in the form of a report to the Commission. On September 1, 1983, the NRC's Office of Investigations issued their interim report on the validity of the employee allegations. OI concluded that many of the allegations related to GPU's internal operations were valid. The NRC's Office of Inspector and Auditor issued a report on September 6, 1983, that concluded that the allegations relating to NRC misconduct were unsubstantiated. The NRC TMI Program Office is currently preparing comments to the OI report.

Advisory Panel/Public Meetings

On July 28, 1983, the Three Mile Island Advisory Panel held a meeting in Harrisburg, PA. Significant items discussed included: Edison Electric Institute (EEI) cleanup funding, DOE cleanup funding, planned underhead characterization studies, NRC investigations into employee allegations, schedule delays with the polar crane load test and head lift, EPICOR/SDS prefilter waste removal, and public perceptions of the Advisory Panel's role in the recovery effort at TMI.

Another meeting was held on August 17, 1983, in Harrisburg, PA, primarily to decide on issues to be presented at a meeting with the NRC Commissioners. At the September 16, 1983 meeting, it was agreed that:

(1) funding the cleanup is the most important issue and that efforts by all should be continued to resolve this problem,

(2) the panel could break into subcommittees, with the latter subject to the Sunshine Act,

(3) the Hartman investigation should be completed as soon as possible,

(4) clarification was needed to determine the safety significance of the OI report.

On September 27, 1983, in Middletown, PA, Mr. Harold Denton, Director of the Office of Nuclear Reactor Regulation, presided over a meeting, which was open to the public, with GPU on the polar crane load test to assess the administration and technical history of the polar crane's refurbishment.

On September 28, 1983, the Advisory Panel held a meeting in Harrisburg, PA. Discussions were focused on levels of funding for 1984 and technical presentations by GPUN and NRC staff on polar crane activities.

TMI-2 Core Examinations

At the end of September 1983, further core examination activities were initiated. These activities include video tape pictures and the taking of grab samples. The results will be analyzed during the fourth calendar quarter of 1983.

* * * * *

83-3 Failure of Automatic Reactor Trip System

This abnormal occurrence was originally reported in NUREG-0090, Vol. 6, No. 1, "Report to Congress on Abnormal Occurrences: January-March 1983." It is updated as follows:

As described in the above report, on February 22 and again on February 25, 1983, the Salem Unit 1 reactor trip breakers failed to trip automatically (and hence, the reactor failed to automatically shutdown) upon receipt of a valid reactor trip signal. On both occasions, the plant was manually tripped a short time after the automatic trip system failed and no fuel damage or release occurred.

After extensive review of the causes of these events and other related issues by the licensee and NRC staff, the Commission, on April 26, 1983, agreed that the Salem Units could return to service. The basis for permitting return to operation was documented in the NRC Restart Authorization Safety Evaluation, which was forwarded to the licensee by a letter dated April 29, 1983 (Ref. B-1). During the review of these events, the licensee made a number of commitments for completing corrective actions, both short-term (before restart) and long-term. On May 6, 1983, the NRC issued an Order confirming these commitments.

On May 20, 1983, after completing all required actions for restart, Unit 1 returned to operation. Subsequently, after completing similar required actions on Salem Unit 2, Unit 2 returned to operation on July 23, 1983 following a refueling outage. NRC Region I inspectors verified that all short-term actions were complete prior to startup of each unit.

As part of the longer term corrective actions, the licensee agreed to engage a consulting firm, Management Analysis Company (MAC), to perform a third party evaluation of the management effectiveness of the licensee's Nuclear Department. The MAC review, the initial results of which were presented to the NRC at the same time that they were presented to the licensee, identified many areas needing improvements but found no deficiencies so serious as to require immediate attention. As a result, the licensee developed and submitted to NRC an Action Plan to address each MAC recommendation and those of other reviews. The Action Plan represents a significant commitment of resources (estimated at 46,000 man-days over the next two years) by the licensee to complete 26 major tasks to improve the operation of the licensee's Nuclear Department.

NRC Region I has also instituted an augmented inspection program to monitor the licensee's progress toward completion of the long-term corrective action program. As part of this program, the NRC staff is reviewing the Action Plan and NRC management is meeting with the licensee every second month to discuss its implementation.

As described in NUREG-0090, Vol. 6, No. 1, on May 5, 1983, the NRC forwarded to the Salem licensee (Public Service Electric and Gas Company) a Notice of Violation and Proposed Imposition of Civil Penalties for $850,000 (Ref. B-2). Violations included operation of the reactor even though the reactor protection system could not be considered operable, and several significant deficiencies which contributed to the inoperability of the reactor trip breakers. The licensee appealed the civil penalty; however, the appeal was rejected by the NRC. Subsequently, the licensee paid the civil penalty on October 28, 1983.

Also as described in NUREG-0090, Vol. 6, No. 1, the NRC Executive Director for Operations directed that a special NRC task force be formed to evaluate the generic implications of the Salem events. The task force's findings were reported in NUREG-1000 (Ref. B-3). Based on information contained in this report, the NRC staff developed required actions related to reactor trip systems reliability and general management capability. These required actions were sent to all reactor licensees on July 8, 1983 by NRC Generic Letter 83-28 (Ref. B-4). The actions covered by this letter fall into the following four areas:

1. Post-Trip Review - This action addresses the program, procedures, and data collection capability to assure that the causes for unscheduled reactor

shutdowns, as well as the response of safety-related equipment, are fully understood prior to plant restart.

2. Equipment Classification and Vendor Interface - This action addresses the programs for assuring that all components necessary for accomplishing required safety-related functions are properly identified in documents, procedures, and information handling systems that are used to control safety-related plant activities. In addition, this action addresses the establishment and maintenance of a program to ensure that vendor information for safety-related components is complete.

3. Post-Maintenance Testing - This action addresses post-maintenance operability testing of safety-related components.

4. Reactor Trip System Reliability Improvements - This action is aimed at assuring that vendor-recommended reactor trip breaker modifications and associated reactor protection system changes are completed in pressurized water reactors (PWRs), that a comprehensive program of preventive maintenance and surveillance testing is implemented for the reactor trip breakers in PWRs, that the shunt trip attachment activates automatically in all PWRs that use circuit breakers in their reactor trip system, and to ensure that on-line functional testing of the reactor trip system is performed on all light water reactors.

Members of the NRC staff met with representatives from Westinghouse at the East Pittsburgh Switchgear Facility as part of the NRC's Vendor Inspection Program. At this meeting, Westinghouse informed the staff of tests conducted in mid-August on the DB-50 reactor trip breakers (DB-50 breakers are used at Salem).

Eight brand new DB-50 breakers were tested by Westinghouse. Four breakers were identical to those used at Salem and the other four breakers had a slightly modified trip pin arrangement. During the testing, some failures occurred, attributed to the lack of 100% lubrication. Westinghouse has issued notices to their customers on the preferred method of applying lubrication. The NRC staff is studying the test results to determine if any immediate action is necessary; also the NRC staff will ensure that licensee programs for breaker preventive maintenance are fully responsive to this issue. It remains to be seen whether the new lubrication technique is as effective under actual plant conditions as compared to testing laboratory conditions.

As a further example of keeping licensees informed of further developments regarding breakers, the NRC issued Inspection and Enforcement Information Notice No. 83-76 on November 2, 1983 (Ref. B-5). The Notice described five malfunctions on October 28 and 31, 1983, at San Onofre Units 2 and 3. The malfunctions involved improper positioning of the undervoltage armatures on the plant's General Electric Type AK-2-25 breakers. The licensees were also alerted that the NRC considers a breaker inoperable if the armature is found in an improper position.

On November 3, 1983 representatives of both General Electric and Westinghouse briefed the Commissioners on various aspects of their breakers. The meeting was open to public attendance and observation. The General Electric representatives described the various types of breakers they supply and emphasized the importance of proper maintenance. Westinghouse representatives presented their views on the cause of the Salem event and actions Westinghouse has taken, including testing of various Westinghouse supplied breakers and development of an improved lubrication procedure.

It is of interest to note that in NUREG-0900, Vol. 6, No. 1, it was stated that based on an independent evaluation of the failed UV trip devices, identified by the licensee, the NRC staff concluded that, while the Salem Unit 1 breaker failures occurred as a result of several possible contributors, the predominant cause was excessive wear accelerated by lack of lubrication and improper maintenance. Westinghouse stated that their observations and confirmatory testing on one of the devices which failed at Salem did not indicate that excessive wear was the cause of failure. They stated that the probable cause of failure was maintenance related.

Both the NRC and Westinghouse observed some wear on several of the undervoltage trip attachments (UVTAs) at Salem. Westinghouse does not characterize this wear as excessive nor do they believe that it would cause the UVTA to fail.

The NRC staff believes that wear of the UVTA can lead to failures via two methods. One method is that wear between specific parts cause the breaker to repeatedly trip free and thus invite tampering to prevent this inconvenience to plant operations. Severe manual deformations of one of the two UVTAs that failed at Salem were observed and reported by both NRC and Westinghouse. This deformed part may have caused the UVTA to malfunction. The other method is that wear leads to greater looseness between parts which increases the chance for binding and/or interference between parts.

The NRC conclusion is based upon the conclusions reached by its consultant, Franklin Research Center (FRC). FRC initially conducted a detailed, independent failure analysis of a non-failed UVTA from Salem. FRC examined additional UVTAs, including one which was traced and verified to have failed at Salem Unit 1. The FRC analyses included electron microscopic and other detailed visual examinations, metallurgical analyses, testing, and review of the operating, maintenance, and surveillance testing history of the breakers. The NRC's consultant's analyses emphasize the importance of proper maintenance and lubrication of the breakers.

The specific and general issues associated with this abnormal occurrence continue to be under active review by the nuclear industry and the NRC. However, unless significantly new issues are identified, this incident is closed for 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 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. Shoreham Emergency Diesel Generator Failures

On August 12, 1983, Emergency Diesel Generator (EDG)-102 at the Shoreham Nuclear Power Plant (99% construction completion) failed due to a fractured crankshaft. The licensee for the plant is Long Island Lighting Company. The plant is a boiling water reactor and is located in Suffolk County, New York. The failure occurred after 1.75 hours of testing at the two-hour overload rating (3900 KW). At the time of failure, EDG-102 had accumulated about 719 operating hours and about 12.5 hours at the two-hour overload rating. The test in progress when the crankshaft fractured was being performed to demonstrate EDG load carrying ability following replacement of all eight cylinder heads with a newer design (originally supplied cylinder heads had developed leaks from the cooling water area). There are three EDG units at Shoreham.

The EDG-102 crankshaft fracture occurred on the generator (load) side of the No. 7 cylinder and extended through the load side crank arm into the crank pin. (The No. 8 cylinder is closest to the load.) Examination of the other two Shoreham EDGs identified cracks similar in location and orientation to the one which developed into a fracture on EDG-102.

The EDGs at Shoreham are Transamerica Delaval Inc. (TDI) Model DSR-48 diesels. Shoreham's EDGs appear to be the only DSR-48 diesels manufactured with a crankshaft assembly having an 11" crank pin diameter and 13" crankshaft diameter. On November 3, 1983, the licensee and its technical consultant reported that the crankshaft failures were definitely caused by a basic design inadequacy. The diesel vendor had previously calculated the average shear stress on the crankshaft to be about 2600 psi. The average stress was actually about 5400 psi and the peak stress was on the order of 66,000 psi. This conclusion is supported by various considerations, which include: industry-standard torsional analysis methods, detailed stress analyses, and actual torsional test results on EDG-101. The licensee is replacing the three 11 x 13 crankshaft assemblies with the 12 x 13 crankshaft assembly design that was reportedly installed in all other DSR-48 diesels. Also, because the Shoreham EDG journal bearings for the crank pins and crankshafts have been identified as being made of out-of-specification materials, those bearings are being replaced.

As a result of the EDG failure and the projected repair and retest time, the Shoreham licensee, Long Island Lighting Company, modified the projected fuel load date from October 1983 until late in the first quarter or early in the second quarter of calendar year 1984. Should the TDI diesels be incapable of being qualified for service at Shoreham, a contingency plan to replace them with higher rated ones from another manufacturer (Colt Industries) is being pursued by the licensee. Such a solution represents a significant but as yet unqualified additional delay in physical readiness for fuel load.

The NRC staff is also concerned about the number and nature of other problems in addition to the crankshaft failures with diesel generators manufactured by TDI. A review of failures experienced at Shoreham and other nuclear facilities raises significant questions regarding the adequacy of the diesel design and manufacturing process. During vendor inspections of TDI which were performed recently by Region IV, at the request of Region I and also in response to allegations or irregularities in the QA program, the staff identified conditions which indicate that portions of the TDI Quality Assurance (QA) Program may not have been carried out in accordance with the provisions of 10 CFR 50, Appendix B. Region IV has referred the QA problems to the Office of Investigations, which has requested that details not be revealed to avoid compromising the investigation. As a result of an inspection performed in July 1983, the staff identified several potential nonconformances with NRC requirements.

The NRC staff will continue to pursue this item at Shoreham and generically. Region I is closely monitoring onsite activities while the NRC Office of Nuclear Reactor Regulation (NRR) is evaluating the generic significance of the Shoreham EDG problems.

The EDG failure at Shoreham was discovered during testing prior to the plant being operational. There was no fuel in the reactor vessel. There was no impact on the public health or safety; therefore, the event is not reportable as an abnormal occurrence.

On August 30, 1983, the NRC issued Inspection and Enforcement Information Notice No. 83-58 to licensees to inform them of the Shoreham event (Ref. C-1). Previous to the Shoreham event, the NRC issued Information Notice No. 83-51 to licensees to inform them of various diesel generator problems (Ref. C-2).

2. Spent Fuel Shipments

For the past several years, most spent fuel shipments have involved portions of spent fuel assemblies being sent to research centers for analysis, fuel being shipped to a government reprocessing center from small research and training reactors, or transfers between two nuclear plants owned by the same utility.

More than ninety percent of the spent fuel now in storage is in spent fuel pools at individual reactor sites. Two non-reactor facilities, however, have also been storing spent fuel (i.e., the General Electric Co. spent fuel storage facility near Morris, Illinois, and the former Nuclear Fuel Services reprocessing plant at West Valley, N.Y.). The West Valley facility stopped receiving fuel in the mid-1970s and General Electric halted receipt of spent fuel in 1980 as a result of an Illinois law which restricted importation of spent fuel into Illinois for storage. The law was subsequently ruled unconstitutional in the federal courts and the decision was left standing by the U.S. Supreme Court in June 1983. General Electric has since begun plans for resumption of shipments to its Morris facility.

The West Valley facility is being decommissioned and as part of the decommissioning effort, the State of New York obtained a federal court order requiring that the spent fuel being stored at West Valley be retrieved by the utilities which had generated it. The return shipments involve four nuclear plants located in Illinois, Wisconsin, New York, and New Jersey.

Accordingly, spent fuel shipments involving the Morris and West Valley facilities have resumed. Shipments began in the third calendar quarter of 1983 and several additional spent fuel shipments are planned. The spent fuel shipments have attracted extensive public attention and interest among state and local government officials.

Seven series of shipments are now underway or in various stages of planning. Four of the series involve the removal of fuel now stored at the West Valley, N.Y., facility for return to the four nuclear plants which generated the spent fuel in the early 1970s. Two series of shipments involve fuel being transferred from two nuclear plants to the spent fuel storage facility at Morris, Illinois. The seventh series involves fuel now stored at Morris being returned to the reactor where it was produced. The shipments are listed in Table C-1.

Spent fuel shipments are under the jurisdiction of the NRC and the U.S. Department of Transportation (DOT). The NRC imposes security requirements and certifies the shipping casks which are used to transport the spent fuel. DOT regulations establish the requirements for radiation safety and highway safety.

The casks carrying the spent fuel are designed to withstand transport conditions including those involved in a severe accident. Because of the accident-resistant nature of the shipping casks, the safety of the shipments rests with the cask and not on any special transportation requirements.

The NRC security requirements include surveys and approval of the shipping route by the NRC (from a security point of view). Additional security requirements, including armed escorts when traveling through heavily populated areas and emergency communications capabilities, are also imposed.

The shipments which began in the summer and fall of 1983 from the Morris and West Valley facilities, respectively, to the Point Beach Nuclear Power Station in Wisconsin have been inspected upon departure and arrival by the persons packaging and receiving the fuel, by NRC personnel and, in some cases, by state personnel. DOT audits the shipments to assure its rquirements are met. Additional surveillance has also been provided by some states through which the shipments are traveling. Ohio, for example, is inspecting the shipments as they arrive in the state and requires a state police escort across the state, at the licensee's expense. Illinois provides surveillance by police and radiation specialists, at the licensee's expense.

Table C-1

Spent Fuel Shipments

Origin/Destination	Status	Route Approved	Weight (Metric Tons)	Number of Fuel Assemblies	Number of Shipments
General Elec. Co. Morris, IL to Point Beach Nuclear Power Station Two Rivers, WI	Underway	Yes	41	109	109
West Valley Nuclear Services West Valley, NY to Point Beach NPS Two Rivers, WI	Underway	Yes	43	114	114
West Valley NS West Valley, NY to Dresden NPS Morris, IL	Underway	Yes	20	206	30
West Valley NS West Valley, NY to Oyster Creek NJ Toms River, NJ	Planned	No	43	224	Not Yet Determined
West Valley NS West Valley, NY to Ginna NPS Ontario, NY	Planned	No	31	81	Not Yet Determined
Cooper NPS Brownville, NB to General Electric Co. Morris, IL	Planned	Yes	211	1056	30
San Onofre NPS San Clemente, CA to General Elec. Co. Morris, IL	Deferred	Yes	31	88	88

REFERENCES
(FOR APPENDICES)

B-1 Letter from D. G. Eisenhut, Director, Division of Licensing, NRC Office of Nuclear Reactor Regulation, to R. A. Uderitz, Vice President - Nuclear, Public Service Electric and Gas Company, transmitting "NRC Safety Evaluation Related to Plant Restart (dated April 28, 1983)," Docket Nos. 50-272 and 50-311, April 29, 1983.* The safety evaluation report is to be published as NUREG-0995.

B-2 Letter from Richard C. DeYoung, Director, NRC Office of Inspection and Enforcement, to Robert Smith, Chairman of the Board, Public Service and Gas Company, transmitting a Notice of Violation and Proposed Imposition of Civil Penalties, Docket Nos. 50-272 and 50-311, May 5, 1983.*

B-3 U.S. Nuclear Regulatory Commission, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant," USNRC Report NUREG-1000, published April 1983.**

B-4 Generic Letter 83-28 from D. G. Eisenhut, Director, Division of Licensing, NRC Office of Nuclear Reactor Regulation, to all licensees of operating reactors, applicants for operating license, and holders of construction permits, transmitting "Required Actions Based on Generic Implications of Salem ATWS Events," July 8, 1983.*

B-5 U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 83-76, "Reactor Trip Breaker Malfunctions (Undervoltage Trip Device on GE Type AK-2-25 Breakers)," November 2, 1983.*

C-1 U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 83-58, "Transamerica Delaval Diesel Generator Crankshaft Failure," August 30, 1983.*

C-2 U.S. Nuclear Regulatory Commission, Inspection and Enforcement Information Notice No. 83-51, "Diesel Generator Events," August 5, 1983.*

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

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

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16. ABSTRACT *(200 words or less)*

Section 208 of the Energy Reorganization Act of 1974 identifies an abnormal occurrence as an unscheduled incident or event which the Nuclear Regulatory Commission determines to be significant from the standpoint of public health or safety and requires a quarterly report of such events to be made to Congress. This report covers the period July 1 to September 30, 1983.

During the report period, there were three abnormal occurrences at the nuclear power plants licensed by the NRC to operate. The first involved large diameter pipe cracking in boiling water reactors; the second involved an uncontrolled leakage of reactor coolant outside primary containment; and the third involved improper control rod manipulations. There were seven abnormal occurrences for the other NRC licensees. Three involved overexposures; two involved medical misadministrations; one involved widespread radiological contamination; and one involved willful violation of license and a material false statement to the NRC. There were no abnormal occurrences reported by the Agreement States.

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

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

BWR Pipe Cracking; Hatch 2 Coolant System Leakage; Hatch 2 Pressure Boundary Degradation; Hatch 2 and Quad Cities 1 Control Rod Misoperations; Overexposures; Willful Violations; Material False Statement; Medical Misadministrations; Radiological Contamination.

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