UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PENNSYLVANIA 19406-1415

March 26, 1998

Russell A. Mellor
Vice President - Operations and Decommissioning
Connecticut Yankee Atomic Power Company
362 Injun Hollow Road
East Hampton, CT 06424-3099

SUBJECT: NRC HISTORICAL REVIEW TEAM REPORT - RADIOLOGICAL CONTROL AND AREA CONTAMINATION ISSUES AT HADDAM NECK

Dear Mr. Mellor:

This letter refers to the special review conducted by various members of the NRC staff from October 1997 through January 1998 of activities authorized by NRC license No. DPR-61 at the Haddam Neck Station, East Haddam, Connecticut. Specifically, the staff reviewed various events and operating practices, dating back to the issuance of your Facility Description and Safety Analysis in 1966, to better assess your radiological characterization and remediation efforts for decommissioning. The team reviewed historical radiological control and area contamination activities that may have resulted in contamination of the site or the environment, the release of radiological effluents, or the insufficient control of licensed material. The team also reviewed your configuration control practices and fuel performance history.

The NRC team's findings and observations are titled, "Haddam Neck Historical Team Report." We are enclosing a copy of the report for your use in the site characterization effort currently being performed by your staff. Your attention is specifically directed to the Executive Summary found on pages two through six of the report. The NRC team has concluded, based on current available information and dose assessments to date, that the conduct of licensed activities at the Haddam Neck Plant over the last 30 years did not result in any apparent radiation exposure to the public or environment in excess of the limits specified in 10 CFR Part 20. The scope and depth of the your staff's efforts to date to review past radiological occurrences and assess significance were found to be appropriate and sufficiently comprehensive for the site radiological characterization. Future NRC inspections will continue to focus on the site radiological characterization activities. In addition, the team concluded that the performance of NRC inspection activities at Haddam Neck, and the application of enforcement, was generally consistent with the agency's existing policy and practices that evolved over time. This team effort has provided the agency with improved information with which to review and evaluate your plans, procedures, and work activities to decommission the facility and affected areas.

Although some items in the report appear to be potential violations of NRC requirements, this report does not address enforcement actions. These items are pending further review and evaluation by the NRC. You will be notified in future correspondence of the results of our review regarding potential enforcement actions.

No reply to this letter is required. However, you may provide comments on any factual statements made in the report that you believe are not correct. If you intend to comment on the report, please notify Marie Miller (610-337-5205) of my staff within 30 days regarding your intent and when we may expect to receive your comments. Your cooperation with us in this matter is appreciated.

In accordance with 10 CFR 2.790 of the Commission's regulations, a copy of this letter and its enclosure will be placed in the NRC Public Document Room.

Sincerely,

A. Randolph Blough, Director
Division of Nuclear Materials Safety

Docket No.: 50-213
License No.: DPR-61

Enclosure: Haddam Neck Historical Review Team Report

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UNITED STATES NUCLEAR REGULATORY COMMISSION

HADDAM NECK

HISTORICAL REVIEW TEAM REPORT

March 1998

HADDAM NECK HISTORICAL REVIEW TEAM REPORT

TABLE OF CONTENTS

APPENDIX B - NRC RESPONSE TO SIGNIFICANT RADIOLOGICAL OCCURRENCES AND OPERATIONAL EVENTS

APPENDIX C - BACKGROUND AND GENERAL REGULATORY PERSPECTIVE

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LIST OF ACRONYMS AND ABBREVIATIONS

ACR - Abnormal Condition Report
AEC - Atomic Energy Commission
ALARA - As Low As Reasonably Achievable
AOR - Abnormal Occurrence Report
B&W - Babcock & Wilcox
BNFL - British Nuclear Fuels, Limited
BOL - Beginning Of Life
BTP - Branch Technical Position
BWST - Borated Water Storage Tank
Ci - Curie(s)
CP - Construction Permit
CY - Connecticut Yankee
CYAPCO - Connecticut Yankee Atomic Power Company
DEI - Dose Equivalent Iodine
ECCS - Emergency Core Cooling System
ECT - Eddy Current Testing
EPA - Environmental Protection Agency
EPRI - Electric Power Research Institute
FDSA - Facility Description and Safety Analysis
FERC - Federal Energy Regulatory Commission
FES - Final Environmental Statement
FSAR - Final Safety Analysis Report
FTOL - Full Term Operating License
GDC - General Design Criteria
ISAP - Independent Safety Assessment Program
kW - kilowatt
LACBWR - LaCrosse Boiling Water Reactor
LCO - Limiting Condition for Operation
LER - Licensee Event Report
LHGR - Linear Heat Generation Rate
LLD - Lower Limit of Detection
LLDP - Low Level and Decommissioning Projects
LOCA - Loss Of Coolant Accident
MCL - Maximum Contaminant Level
MTU - Metric Ton Uranium
MWD - Mega Watt Day
MWe - Mega Watt electric
MWt - Mega Watt thermal
NEPA - National Environmental Policy Act
NMSS - Nuclear Material Safeguards and Safety
NRC - Nuclear Regulatory Commission
NU - Northeast Utilities

ODCM - Offsite Dose Calculation Manual
PAB - Primary Auxiliary Building
PCP - Process Control Program
PCT - Peak Cladding Temperature
PDCR - Plant Design Change Request
PIR - Plant Information Report
PN - Preliminary Notification
POL - Provisional Operating License
PORC - Plant Operating Review Committee
PSDAR - Post Shutdown Decommissioning Activities Report
QA - Quality Assurance
RCA - Radiation Control Area or Radiologically Controlled Area
RCS - Reactor Coolant System
RETS - Radiological Effluent Technical Specifications
RG - Regulatory Guide
RWST - Refueling Water Storage Tank
SALP - Systematic Assessment of Licensee Performance
SDMP - Site Decommissioning Management Plan
SEP - Systematic Evaluation Program
SER - Safety Evaluation Report
SGTR - Steam Generator Tube Rupture
TLD - Thermoluminescent Dosimeter
TS - Technical Specifications
UFSAR - Updated Final Safety Analysis Report
USQ - Unresolved Safety Question
UT - Ultrasonic Testing
XPE - Xenon Pin Equivalent

NRC TEAM REPORT - HISTORICAL REVIEW OF HADDAM NECK

A. Objectives

The objectives of this review were to: (1) gain better understanding and appreciation of the scope and extent of previous radiological occurrences in order for the NRC to better assess the acceptability of the licensee's future site radiological characterization efforts and subsequent remediation of affected areas, on-site and in the environment; and (2) identify whether licensee activities that resulted in contamination of the site, uncontrolled or unmonitored effluent releases, or insufficient control of licensed materials were considered for (or subject to) action relative to existing NRC regulatory requirements, including enforcement.

While sufficient for its purpose, this effort was not intended to be an exhaustive study and review of every contamination event and circumstance that occurred within the 30-year operational period of the Haddam Neck plant. Nor was it intended as a comprehensive examination and assessment of every regulatory action, document or record that might have pertinence. This report is not a substitute for the licensee's historical assessment being conducted as part of the site characterization. Rather, this effort was designed to provide understanding, clarification and perspective of licensee practices that resulted in facility contamination and certain significant events or conditions that had the potential to affect public health and safety or impact the environment. Accordingly, the NRC team selected and examined events and circumstances that appeared to be most significant and provided the best insight into Haddam Neck's past performance regarding radiological control, along with the NRC's corresponding oversight.

B. Approach

To accomplish these objectives, the NRC team reviewed documentation pertaining to licensee performance and NRC regulatory activities over the operating period of the plant relative to the stated objectives. Documents from 1966 to 1997, were reviewed at the Haddam Neck site, at NRC Region I, and at NRC Headquarters. NRC Regulations and radiation detection technology evolved over that period, which required the NRC team to review events in context with the regulations and technology in existence at the time. The NRC team's findings and observations for each objective are documented in separate Appendices to this report.

Appendix A, "Review of Licensing/Design, Processes and Events That Led to Radiological Occurrences," describes findings and observations regarding: (1) the licensee's historical review of events and circumstances that led to certain radiological occurrences that affected the radiological status of the site; (2) offsite contamination as a result of licensee practices; (3) the process for monitoring and controlling the release of radioactively contaminated materials from the site; (4) the licensee's documented radiological environmental monitoring reports; (5) the licensing basis and operating experience associated with radioactive waste processing; (6) the process and practices for monitoring and controlling non-radiological system and release pathways that became contaminated due to events or licensee practices; and (7) the licensee's experience with stainless steel clad fuel, and the events and circumstances that resulted in fuel clad defects. Appendix A also includes supplementary information, having pertinence to these findings, such as copies of licensee's initial scoping survey maps, for further clarification and understanding.

Appendix B, "NRC Response to Radiological Occurrences and Events," describes past AEC and NRC inspection and enforcement response regarding the circumstances and conditions that resulted in various contamination events. These details include observations and findings regarding the licensee's efforts to report events, the NRC response to significant events and review of the inspection record, including enforcement action. For perspective, information on the scope and extent of enforcement, in other regions and on similar issues, is described. Supplementary information, i.e., chronological summary of events and NRC response, and listing of NRC enforcement actions specific to Haddam Neck, is provided.

Appendix C, " Background and General Regulatory Perspective," describes the emerging radiological control performance issues that led NRC to establish an action plan to perform a historical review of the radiological control and area contamination issues at Haddam Neck. A limited discussion on the regulatory functions of NRC and its development over time is also included to provide a perspective to the team's historical review.

C. Executive Summary

Based on currently available information and dose assessments to date, the conduct of licensed activities at the Haddam Neck Plant over the last 30 years apparently did not result in any exposure to the public or environment in excess of the limits specified in 10 CFR 20. However, recent findings from the licensee's historical survey efforts have identified a radiation program breakdown in 1975 that resulted in the inappropriate release of contaminated concrete blocks for unrestricted use. While there is potential for public exposure in excess of 10 CFR 20 limits, based on observed use and condition of these blocks, there has been no evidence of such exposure, to date. The final determination of this matter will require further radiological surveys and additional assessment of the historical use and condition of the blocks.

Operation of the Haddam Neck facility resulted in various spills, leaks, and unplanned effluent release of radioactive materials. There is no evidence that plant operations resulted in the licensee exceeding any public exposure regulatory requirement as specified in 10 CFR 20. However, because of the fuel cladding defects in 1989, there was an instance in which the safety Technical Specification Limit of 10 millirad for the quarterly beta air dose was exceeded. The calculated dose from that event was 11.9 millirad to a hypothetical person at the protected area boundary. In all cases observed, there was no significant radiological consequence to public health and safety.

Most spills and leaks of radioactive materials appeared to have been confined to the Radiological Controlled Area (RCA). The licensee subsequently performed limited remediation to prevent or limit the spread of the contamination. In accordance with licensee procedures, the material was either disposed of at a low-level waste facility or released for unrestricted use. In addition to the concrete blocks mentioned above, recent findings indicate that some soil and debris, containing low level or trace concentrations of licensed material, were inappropriately released for unrestricted use. The NRC team determined that the circumstances in these cases generally involved either: (1) the licensee's improper application of the limits specified in 10 CFR 30, Schedule A and B (Exempt concentrations and quantities), and 10 CFR 20, Appendix B (Effluent concentrations), as unrestricted use release criteria, or (2) the licensee's failure to maintain effective oversight and control of contaminated materials (e.g., concrete blocks) that were known or suspected of being contaminated. These apparent performance deficiencies were not identified until site characterization efforts were initiated in 1997 during preparation for decommissioning. Subsequent confirmatory measurements and radiological assessments by the licensee, the State of Connecticut's Department of Environmental Protection and the NRC, to date, have not revealed any contamination in any off-site location that currently would exceed 10 CFR 20 limits. Accordingly, the impact on public health and safety is not expected to be significant. However, final surveys and dose assessments have not been completed.

Another factor pertinent to the release of materials from power reactor sites for unrestricted use is the effect of improvements in radiation detection technology. The regulations in 10 CFR 20 governing the disposition of radioactive materials require that any detected activity must be dispositioned in accordance with NRC requirements. However, the sensitive laboratory methods now available permit the application of a lower limit of detection than was reasonably achievable in earlier times. Therefore, it is now possible to detect trace activity in materials that may have been adequately monitored and released in accordance with the existing guidance of that time.

The NRC team reviewed records of the licensing basis for the following: 1) handling of radioactive materials and radwaste processing, along with actual licensee practices, and 2) monitoring of fuel performance over the time period of interest. The licensee's configuration control practices contributed to inadvertent releases from the waste gas decay tank and spent fuel building floor drain. These included a modification of the radioactive waste processing system in 1975 that was not adequately evaluated by the licensee in accordance with 10 CFR 50.59, as well as the conduct of radioactive waste handling activities, in 1989, in the spent fuel building that was not described or reviewed by a safety evaluation. The licensee experienced throughwall fuel cladding defects in 1979 and 1989 that resulted in licensing action. While affecting the radiological condition of the RCA and areas within the licensee's protected area, none of these situations resulted in any circumstances that would be expected to cause significant health and safety consequences relative to the public. Some of these conditions may be potential violations of agency requirements that were not previously identified for enforcement action. However, the doses to workers and the public resulting from these situations were within the requirements of 10 CFR 20. These apparent violations will be further reviewed by the NRC staff and considered for enforcement actions in accordance with the NRC Enforcement Policy (see Appendix B, Section 2.3).

Tritium from routine effluents and from mid-1970's leaks in the underground liquid waste test tank lines resulted in onsite groundwater contamination and measurable concentrations in the Connecticut River. Because characterization of the tritium plume has recently begun, it may be possible that higher concentrations could be detected as well as the identification of other contributing sources. A selected review was performed of the licensee's Annual Radiological Environmental Reports. As required, the licensee reported tritium in groundwater and the Connecticut River. Dose consequences to the public were within the limits of 10 CFR Part 20 and the EPA Maximum Contaminant Level established in 1976. Independent environmental monitoring by the State of Connecticut was in agreement with the licensee's data.

The team determined that the licensee's formal event notifications were generally in accordance with the requirements of 10 CFR 50.73 and 10 CFR 50.72. Exceptions included a late and incomplete notification of the fuel defects in 1989 and the 1997 discovery of contamination that had been released from the RCA and disposed of in a landfill within the owner-controlled area, which was accessible to the public. Regarding the 1989 fuel defects, this event resulted in the plant exceeding a design basis limit (1% failed fuel assumed in the waste gas decay tank rupture accident), but apparently the licensee did not recognize or report this event as such.

Previous NRC inspection activities were generally conducted in line with agency rules and regulations in effect at the time. Over the last 30 years, NRC inspection reports documented the agency's reviews of plant programs that included radiation protection, radiological controls, radiological waste processing, and effluent and environmental monitoring. The staff weighed a number of factors in deciding on the nature, extent and timing of NRC follow-up to events at the plant. These included the apparent safety significance, the general performance of the plant operator in the area involved, and competing inspection priorities. The team did not attempt to reconstruct the factors or competing inspection demands that influenced the staff's response and follow-up to past events or occurrences. Nevertheless, the team was able to conclude that for most radiological events, NRC follow-up was commensurate with the expected safety impact and was, therefore, appropriate. Spills and releases typically involved limited contamination and did not result in appreciable dose to workers, the public, or environmental impact, and did not effect operation of the facility. The licensee generally conducted remediation of spills and contamination occurrences in owner controlled areas when contamination was identified. While NRC did not always examine each individual occurrence, normal inspection program activities were sufficient to verify that remaining residual contamination would not result in radiological exposure to workers or the public in excess of regulatory requirements. In general, the agency's focus in radiological control inspections was principally on assuring that the licensee's programs for environmental and effluent monitoring, and radiation protection and radiological control were maintained in conformance with regulatory requirements. Further, the priority for both the plant operator and the NRC inspection program was on the immediate control of radiation exposures to workers and the public, and generally did not consider the potential affect of site contamination events on future decommissioning, including financial impact. These priorities may have led to limited assessment of some individual radiological events. Under these circumstances, there were possible missed opportunities to gain performance insights that may have affected the NRC's assessment of the plant operator's overall performance and consideration of possible enforcement action.

Enforcement action was usually taken for operational problems that were considered safety significant, but not for small spills and releases, because of the negligible dose impact to plant workers or the public. Further, these smaller events did not contaminate areas outside the protected area which was consistent with the licensing basis of the facility. Therefore, the application of enforcement action varied with the specific circumstances and the safety significance of each event. It is apparent that there were a few missed opportunities where NRC should have taken enforcement action in the past. These items will be further reviewed and evaluated using the NRC enforcement policy to determine any actions that may be taken.

D. Conclusions

The scope and depth of the licensee's current effort to review past radiological occurrences and assess significance are appropriate and sufficiently comprehensive for the site radiological characterization, as required by 10 CFR 50.82(a)(9)(ii). This was determined by the team's review of the licensee's initial scoping efforts for site characterization, as documented in the following: radiological classification of plant systems and land areas; surveys and reports of past operational occurrences; procedure review for releasing materials from the facility for unrestricted use; and licensee interviews with personnel and members of the public, who acquired materials from Haddam Neck. The licensee's continuing efforts to finalize the site radiological characterization will be a focus of future NRC inspections.

Over the 30-year operating period of the plant, there were several occurrences and events that resulted in contamination of the facility and the immediate environment. Fuel clad defects led to increased radiological source term and deposition of transuranic activity in radiological and non-radiological plant systems. In 1979, while operating with an increased transuranic source term in the primary system, Haddam Neck experienced several inadvertent liquid and gaseous releases. The contamination outside the RCA from these events was not discovered by the licensee for several months. Isolated spots were found in the protected area and at the parking lot within the owner controlled areas. No significant impacts were identified by the licensee's environmental monitoring program. Although remediation of identified areas was completed in 1980, recent scoping surveys of the hillside have identified some small spots with transuranic and other fission product activity. Because of the radiological waste filtration and clean-up systems, most spills and releases to the environment that occurred did not impact areas outside the owner-controlled property. However, tritium entered the environment through routine effluent releases and system leakage. These conditions were within regulatory requirements.

Recent revelations of low-level or trace concentrations (quantities) of licensed materials in some off-site locations provide evidence of previous deficiencies in licensee procedures or performance with respect to radioactive material control. Subsequent off-site confirmatory measurements and assessment of the existing conditions by the licensee, the NRC and the State of Connecticut-Department of Environmental Protection have not revealed any radiological concentrations or subsequent exposures significant to public health and safety, to date. Evaluations and assessments are still in progress.

The NRC team noted that the current technological capability permits the application of a lower limit of detection than was reasonably achievable in earlier times. As such, it is now possible to detect trace activity in materials that were effectively monitored and released in accordance with the existing guidance at that time.

The performance of NRC inspection activities at Haddam Neck, and the application of enforcement, was generally consistent with the existing policy of the NRC and practices that evolved over time. Major operational events and larger spills or releases were typically reviewed and considered for potential enforcement actions. Events that were expected to have minimal impact on workers, the public, or the environment received limited NRC review and follow up that was consistent with inspection priorities. Notwithstanding, it is apparent that there were a few opportunities for the agency to more rigorously review events or situations to determine the appropriate enforcement actions. These items will require further review and consideration in accordance with the NRC enforcement policy. This review will consider the relationship of the issues to the current licensed activities and the need for corrective action to prevent recurrence.

This team effort has provided the agency with better information with which to review and evaluate licensee plans, procedures and work to decommission the facility and remediate affected areas.

APPENDIX A

REVIEW OF LICENSING/DESIGN, PROCESSES AND EVENTS THAT LED TO RADIOLOGICAL OCCURRENCES

1. LICENSEE'S HISTORICAL REVIEW

Scope

The licensee's preliminary historical review records and scoping survey plans were reviewed by the team to gain an understanding of the scope and extent of previous radiological occurrences at the Haddam Neck site. The NRC team developed information from licensee documentation of surveys and reports of spills and releases, as well as from results of licensee interviews with personnel. It is expected that this information will permit the NRC to better assess the licensee's site characterization and remediation efforts, and to determine the acceptability of the licensee's termination plan, as required by 10 CFR 50.82.

During the course of plant operation, radiological conditions developed in the Radiological Controlled Area (RCA) as the result of the processing and handling of radioactive waste and effluents. In some cases, areas in the RCA became contaminated. Though remedial action was taken by the licensee, residual contamination occasionally migrated from the RCA into the surrounding owner-controlled property. In other cases, events involving gaseous effluent releases may have deposited materials outside of the RCA. Also, the licensee's process for release of material from the RCA to unrestricted areas was not adequate. The review included a selected examination of the licensee's identification, assessment and follow-up actions for these situations.

1.1 Site Characterization

The purpose of site characterization is to identify the type, location and concentrations of contamination present on the Haddam Neck site in order to determine what remediation is necessary to decommission the facility. This information is used to estimate the volume and class of waste material, by evaluating the radioactive contamination of the land areas, systems and structures of the facility. Besides the decommissioning planning, it also supports the final status survey process by identifying the areas that may require more monitoring and sampling. NRC requires, through 10 CFR 50.75(g), that licensees keep records of information important to the safe and effective decommissioning of the facility in an identified location, until the license is terminated.

The regulation requires the licensee to maintain records of spills or unusual occurrences that result in significant contamination remaining after remediation efforts. In such cases, the licensee must implement adequate radiological controls to assure regulatory requirements are maintained. Provided that all regulatory requirements can be maintained, the licensee is not required by the regulations to fully remediate contaminated areas on its property to background levels. However, the contaminated areas must remain under the control of the licensee until released in accordance with regulatory requirements.

In early 1997, the licensee initiated a radiological characterization scoping survey which included survey and identification of potentially contaminated areas inside and outside the RCA. As part of this review, the licensee examined previous Adverse Condition Reports (ACRs), a problem identification and corrective action reporting system. On June 30, 1997, the licensee identified that the 10 CFR 50.75(g) decommissioning record file was not completely current and did not contain all the information required by 10 CFR 50.75(g). The licensee reported this discrepancy in ACR 97-0381. Subsequently, the licensee undertook an historical review to recreate this file, in conjunction with on-site scoping surveys, to identify the extent of on-site contamination in suspected contaminated systems and land areas.

The team noted that the licensee had initiated a 10 CFR 50.75(g) file in 1990 by compiling a list of previous events reported to NRC and from the licensee's Plant Incident Reporting System (now ACR system). However, as identified by the licensee in 1997, this file was incomplete and had not been maintained. The NRC inspection record does not indicate that the requirements of 10 CFR 50.75(g) were inspected. Further, areas outside the radiological controlled areas were not included in the licensee's file until they were identified by the scoping surveys conducted from July through October 1997. Such areas included the landfill area (shooting range); the hillside behind the fuel building (east side of the site); the storm drain run-off area south of the site; and the peninsula area (southwest storage area).

The licensee documented radiological surveys of the plant site starting in 1967. Dose rate surveys were performed quarterly for the perimeter and areas within the radiological controlled area boundary. In 1979, the licensee began an annual site survey which included areas outside the radiological controlled area boundary but within the owner controlled area. The surveys in 1979 included monitoring for loose (removable) contamination in addition to the dose rate measurements.

1.2 On-site Contamination

The licensee's site characterization effort also involved the identification of significant on-site contamination events. As of September 19, 1997, the licensee had documented or identified approximately 125 individual "events" (e.g., an activity, event or spill) that may have resulted in residual contamination of the site over its operating history (see Supplement A-1 to this Report). Of the 125 events, about 12 involved non-radiological type events (e.g., oil spill). The events, dating back through 1969, were documented in records such as abnormal occurrence reports, plant incident reports, licensee event reports, adverse condition reports and event notifications. In general, each event included an event description and a statement of what corrective action (including remediation efforts) was known to have been taken at that time. The licensee estimated that complete information (quantities of materials, drawings, documentation of remediation actions and survey records) was only available for approximately 10 percent of these events.

The licensee has performed (and is continuing to perform) radiation surveys of the site to document the type and levels of radioactive material present. One licensee report reviewed was the "Investigation of the Source of the Radioactive Contamination Found on the Connecticut Yankee Site March 10-30, 1980," dated April 1980. This report documented the results of extensive radiological surveys performed on plant buildings and site property. The surveys revealed the presence of licensed radioactive material in areas beyond the radiological controlled areas. The licensee identified and remediated areas where the radiation levels were above NRC limits for non-radiologically controlled areas. This information was reported to the NRC and the State of Connecticut Department of Environmental Protection within one hour of discovery. The licensee believed the likely source of the contamination resulted from the release of radioactive material through the primary vent stack after actuation of the degasifier rupture disk in February 1979, and again in December 1979, and possibly from residue from the cleaning of the stack in September 1979. The licensee performed a dose assessment which assumed the radioactive material was transported offsite and exposed a member of the public, and that the exposure was averaged over the entire skin of the whole body. The calculated potential doses to the skin and to the gastrointestinal tract were 0.75 mrem and 0.3 mrem, respectively. These calculated doses are a very small fraction of 10 CFR Part 20 annual dose limits and within the ALARA criteria of Appendix I to 10 CFR Part 50. However, the skin dose, when calculated over 1 square centimeter, which is consistent with regulatory guidance, could have approached the occupational quarterly limit for skin of the whole body (7.5 rem/quarter). Because these discrete spots of contamination were not widespread, the likelihood that a person would have received a skin contamination in excess of the occupational skin limit is remote.

The licensee's report also identified residual levels of radioactive material in mud sediments along a storm drain runoff leading from the facility grounds to the discharge canal. The licensee identified that the contamination likely resulted from the discharge of contaminated liquid from the storm drain which originated within the radiologically controlled area and from runoff from the protected area. The contaminated runoff likely originated from contamination on the ground, which resulted from leaking radioactive liquid storage tanks and from radioactive waste handling operations in the outside environment but within the radiologically controlled area.

This radioactive material from the storm drain and the runoff represented an unmonitored release pathway. There was no barrier to prevent the radioactive material from migrating into the licensee's discharge canal and being transported into the unrestricted area. Because the radioactive material released into the discharge canal through this pathway was not monitored, the licensee did not have data to support compliance with NRC regulations.

The licensee's documentation indicated that areas of potential residual contamination were principally located around (and potentially under) various radiological controlled process buildings. A site map depicting suspected areas of residual contamination is included as Supplement A-2. These buildings are centrally located on the site and within the protected area. The licensee performed core bores at three locations around this area and noted that, based on these limited preliminary samples, no significant subsurface (e.g., greater than 6 inches) residual activity was present. Additional samples are planned.

Areas where residual contamination may be present included locations previously used for outdoor handling of radioactive waste (e.g., outdoor resin handling station). Other suspected locations include an area known as "the ballfield" (an area within the protected area) and an outfall area at the south end of the facility (outside the protected area but within the owner controlled area) known as the "leachfield."

The "ballfield" may have received potentially low-level contaminated fill soil from building excavation projects when it was paved over. The NRC team noted that an apparent excavation was performed to support the construction of the radwaste reduction facility, as described in plant design change request No. 85-733, dated October 15, 1985. In the course of the excavation, the licensee detected low-level soil contamination, excavated the contaminated soil to a pre-determined specific activity based on an evaluation, disposed of the material by transfer to a licensed disposal site and performed a dose calculation based on residual radioactivity remaining in the excavation. The licensee performed direct frisking of soil and analyzed it using gamma spectroscopy. Existing records indicated that any remaining contaminated soil was drummed and disposed of as radioactive waste. As discussed in Section 3 of this report, the licensee's release criteria were inappropriate, at that time, and may have resulted in the release of small concentrations of radioactive material.

In July 1997, the licensee became aware that an area known as the "landfill," located about 0.25 miles northeast of the station in the owner-controlled area, had received fill/rubble from a previous on-site work activity. The licensee performed radiological measurements at the location and detected low-level concentrations of radioactive material in the soil. Cobalt (Co-60) activity in isolated spots ranged from about 0.31 pCi/g to 4.3 pCi/g. Cesium (Cs-137) ranged from 0.17 pCi/g to 34.8 pCi/g. The licensee collected material from the area (e.g., fabric, soil, brick) that indicated 400 - 600 corrected counts per minute (ccpm). The licensee performed preliminary estimates of potential doses to personnel who may have inadvertently entered the area and concluded that any dose received would be well within NRC regulatory limits. The area was subsequently fenced in and designated as a radiological controlled area pending further evaluation.

The southwest site storage area (also known as the "boneyard") is located on the peninsula between the discharge canal and the Connecticut River. The area was used as a storage area for various items throughout the operation of the facility, including potentially radioactive/contaminated items. In addition, other portions of the peninsula were used for storage of dredged material from the discharge canal. As part of the site characterization, the licensee reviewed records to determine whether the area was surveyed periodically for radioactive materials. A survey performed by the licensee in March 1980 revealed a section of concrete slab with dose rates up to 500 millirem/hour on contact. The slab was buried under approximately 1.5 feet of soil. After removal of the concrete, there were no other areas found with elevated dose rates. A licensee investigation revealed that radioactive material had been stored in this location and the contamination could have been inadvertently left in the area when the material was removed.

Of the 125 radiological occurrences identified by the licensee, most did not result in any significant contamination. The following radiological occurrences resulted in some level of site contamination that may require further remediation to support decommissioning:

- Leak from the radioactive water storage tank (RWST) heater valve in November 1973 that contaminated the storm drain system;

- Multiple waste gas tank rupture disc actuations in the 1970s;

- Various leaks in the steam generator blowdown waste discharge line and the service water effluent line under the Primary Auxiliary Building (PAB) floor in the 1976-1980 time period;

- Contamination of the yard area around the Borated Water Storage Tank (BWST) from a leak in the circulating water heater line in 1978;

- Unplanned radioactive release from the degasifier through the plant stack in December 1979;

- Leak from a cracked weld seam in the auxiliary building exhaust duct to the main stack in September 1981;

- Resin liner overflows in 1984;

- Dredging of the discharge canal in 1986;

- Drain hose spill of contaminated water to the yard area in August 1987;

- Contaminated water from radioactive waste processing dumped into an uncontrolled drain that emptied into the 115 kV switchyard trench in February 1989;

- Spill of component cooling water to the storm sewer in March 1990;

- Leak from the Refueling Water Storage Tank in September 1990;

- Spill from the Reactor Coolant System to the pipe trench in August 1991; and

- Draining of the PAB heat exchanger to an uncontrolled drain that emptied into the 115 kV switchyard trench in April 1984.

Routine operations led to contamination of groundwater at the Haddam Neck site. However, the amount of contamination in groundwater leaving the site is limited by Appendix B of 10 CFR 20. NRC does not have a general regulatory position or guidance on groundwater monitoring at Part 30, 40, 50, and 70 non-waste disposal facilities. If any groundwater monitoring is performed at these sites, it is done through license specific requirements. Although not specifically provided to reactor licensees, the NRC's Nuclear Material Safeguards and Safety (NMSS) Low Level Waste and Decommissioning Projects (LLDP) Branch did publish, and notice in the Federal Register, a Branch Technical Position (BTP) entitled "When To Remediate Inadvertent Contamination of the Terrestrial Environment" in October 1994. This BTP recommends to licensees that known or suspected releases to groundwater need to be characterized, and remediated as appropriate, as soon as possible. Timely remediation would minimize health and safety problems. The continued presence and movement of contaminated soil and/or groundwater over time could also increase the volume of contaminated material and therefore increase the cost of decommissioning. However, by regulation, power reactor licensees are not required to remediate areas that are inaccessible until decommissioning.

Tritium from routine effluents and spills is present in the groundwater on-site. Tritium is highly mobile in the environment and is easily detected in groundwater samples after a release. The groundwater contamination at Haddam Neck was monitored in the radiologically controlled area at the external containment sump (ECS) near the containment building, and outside the radiologically controlled area, but on the owner-controlled property, at two water supply wells adjacent to the discharge canal. The primary source of the tritium was identified as coming from waste test tanks. The Haddam Neck staff first identified the source of the contamination in the sump in 1976. The source of the tritium in the ECS was suspected to be due to leakage from monitoring tanks. The source of the tritium in the wells was suspected to be due to migration of tritiated water in the discharge canal to the local aquifer penetrated by the wells. Because the water from the two wells was a nonpotable water source for the facility, with tritium concentrations above background, this could have been an unmonitored dose pathway, and it might not have been within the principles of As Low As Reasonably Achievable (ALARA) to use this water for any domestic purposes at the site. Although the on-site well water was used for process water, there were no controls to prevent facility workers from drinking the tap water. However, the dose calculations in NRC Inspection report 50-213/97-11 indicate that the potential doses from tritium, even if the water was used as a drinking water source, would have been low (< 1 mrem/yr) and not a health and safety concern.

EPA Interim Drinking Water regulations in 1976 established a maximum contaminant level (MCL) of 20,000 pCi/l for tritium. These were written for public drinking water supplies serving 25 or more people. The drinking water MCL does not apply to Haddam Neck's use of the groundwater. EPA's CERCLA program guidance requires the application of MCLs in the groundwater plume that is a current or potential source of drinking water. In addition, the NRC's 1992 Site Decommissioning Management Plan (SDMP) Action Plan for decommissioning suggests that MCLs be used as reference standards for groundwater remediation at decommissioning sites. However, the recently promulgated decommissioning criteria rule specifically excludes using MCLs as a separate standard for groundwater contamination at decommissioning sites. NRC is aware of several NRC licensees that have contaminated groundwater on-site. The fact that a licensee has contaminated the groundwater at its site (above MCLs in some cases) is not a specific violation of NRC regulations unless the regulatory requirements of 10 CFR 20 or license conditions are exceeded. Notwithstanding, the potential dose from any groundwater radioactive plumes would be determined during site characterization. The evaluation would determine if groundwater remediation would be required.

Haddam Neck staff kept records of the groundwater tritium concentrations over the years, and while the source of the contamination in the sump was believed to be known and remediated, there continued to be tritium in the sump water at varying concentrations. This is somewhat problematic as other sources of tritium may have been contributing to the concentrations found in the sump over the years, but their presence would have been masked by the previous contamination and the assumption that the previous contamination was the source of all subsequent tritium levels measured in the sump. Apparently, no attempt was made to characterize the groundwater plume that was contributing tritium to the sump. It is possible tritium in the groundwater could be coming from an unidentified source. In addition, without characterizing the plume there is the potential that the measuring point(s) are not in the correct location to detect the maximum concentration of the plume.

Groundwater at Haddam Neck flows into the Connecticut River, which is not a drinking water source, downstream of the site. Thus, dose to the pubic via the drinking water pathway is essentially zero.

Conclusion

The scope and depth of the licensee's current effort to review and document the site's history regarding contamination events and radiological occurrences are appropriate and sufficiently comprehensive. The licensee has identified over 125 events, some of which contributed to the current radiological condition of the facility, that could have an impact on the decommissioning. While these events resulted in the potential for, or the occurrence of, radioactive materials being released outside the confines of the RCA, the licensee's radiation survey of the site and evaluation does not reveal any evidence that the quantities or concentrations represented a significant radiological hazard to plant workers, members of the public or the environment. Where the radiation levels exceeded NRC regulations or reporting criteria, the licensee made the appropriate reports and remediated the areas.

NRC regulation 10 CFR 50.75(g) requires recordkeeping of spills or other unusual occurrences involving the spread of contamination in and around the site. However, the records may be limited to instances when significant contamination remains after any cleanup is done. The regulation does not require remediation to background radiation levels. The records of the contamination and its location will be used to decommission the site. Prior to the 1990 effective date of the regulation, the licensee was not required to have specific records on contaminated areas to facilitate the ultimate decommissioning, except for records related to on-site waste disposals. Routine surveys of the radiological controlled areas of the facility would have been performed to demonstrate compliance with the radiation standards in 10 CFR Part 20. For the significant contaminated areas identified by the licensee at Haddam Neck, complete records in accordance with 10 CFR 50.75(g) were not available to the team during the review. NRC Inspection Report No. 50-213/97-08 identified this as an unresolved item.

The tritium concentrations, although below the MCLs, are an indication of previous or current leakage from systems that contain radioactive materials. Tritium is highly mobile in the environment, so it is easily detected in groundwater after a release. Other non-soluble radioactive contaminates would not normally be expected to be detected in groundwater. The tritium monitored at Haddam Neck could indicate that some soil near the original spill or release point may require remediation. However, dose to the public via the drinking water pathway is essentially zero. Characterization of the tritium plume has been initiated as part of the licensee's site characterization.

2. OFF-SITE CONTAMINATION

Scope

This section reviewed the recent efforts by the licensee to characterize the quantity and concentration of materials that were released from the site. The licensee's offsite characterization has not been completed.

General requirements for disposition of licensed materials are listed in 10 CFR 20.2001. Because this regulation does not define an exempt quantity, any amount of detected licensed material must be dispositioned in accordance with NRC requirements. By using sensitive laboratory methods, trace amounts of licensed material may be detected at levels considerably less than the Lower Limit of Detection specified through other regulatory guidance or requirements. At such levels, there is no expected impact on public health and safety.

Details

2.1 Offsite Soil Releases

Various property owners informed the licensee that they had received soil from the plant, along with general fill material (asphalt, concrete, soil) during plant construction projects in the 1980s and 1990s. The fill was excavated from the site when CYAPCo constructed new buildings on-site (such as the emergency operations facility, the switchgear building and the radwaste reduction facility) and renovated a parking lot on the north side of the site. Although most excavated materials were taken to the licensee's landfill area on the south side of the site, a considerable amount of material also was released to the public for unrestricted use. The fill materials came from both inside and outside the radiological control area at the plant.

The licensee identified about 12 offsite areas that were believed, with reasonable assurance, to have received some fill/rubble from the site. These areas were identified by direct contact with various local property owners and by public response to notifications, press releases and media reports on the matter. Areas to be investigated were assigned to a matrix to positively identify the area for follow-up and to develop information on the time and circumstances under which the materials were received. The licensee initiated a walkdown of the subject properties to identify the areas potentially affected by plant-related materials. The results of the site walkdown were used to develop a specific survey and soil sampling plan of the suspect areas at each location.

The licensee also conducted similar surveys and soil sampling of areas on-site that were open for unrestricted public access, such as the north parking lot, the picnic areas, the boat launch access area and the nature trails on the north and east side of the plant.

2.2 Contaminated Blocks from Shield Wall

As part of the decommissioning process, the licensee's historical site survey identified that material from radiologically suspect areas of the plant had been released off-site. Specifically, based on interviews with plant workers who received the material, the licensee identified that workers took possession of solid cement blocks following the demolition of a wall in 1975. The blocks had been used as a shield wall around a former cask washdown pad that presently is the location of Bus 10. The licensee estimated that 5,130 blocks were used to construct the wall. The blocks measure 4" X 8" X 16".

In the early 1970s, the cask pad was used for temporary storage of contaminated filters, resin liners and trash. At least one liner leaked. The leakage contaminated the storage area, including some of the blocks which contained it. Once the failed liner began to dry, airborne radioactivity was identified in the area. One worker recalled that remnants of the failed liner had contact levels of 10 R/hr.

After abandoning use of the cask pad as a storage area, the licensee dismantled the wall and began a process to survey the blocks to separate the contaminated ones from those unaffected by the contamination, with the intent to release the uncontaminated blocks. Plant workers were allowed to take blocks directly from the partially dismantled shield wall and to frisk the blocks for free release. When interviewed in 1997, most workers did not remember the type of survey instrument(s) used, or the release criteria that applied. While workers stated they checked the blocks for radioactivity, it was not certain that every worker checked each block. Health physics technicians helped some workers check blocks for contamination. Some workers, who were qualified in radiological controls, surveyed blocks during work shifts and separated blocks to be released into piles for each worker. The workers loaded the blocks into a truck at the end of a shift and removed them from the site. Based on entries in a security gate log, the process of frisking and taking possession of blocks occurred over the period of September through November 1975. Several workers took many truckloads of blocks. Subsequently, the blocks were used to build structures, walkways, ramps, retaining walls and landscaping borders. Some blocks were used inside the home (i.e., cellar).

In late 1997, the licensee issued a Licensee Event Report (LER) to the NRC regarding the breakdown in the health physics program that led to the release of contaminated material from the site. The licensee initiated corrective actions, including the survey, evaluation and removal or disposal of contaminated materials. NRC inspectors have performed independent measurements and analyzed split samples with the State of Connecticut and the licensee. The preliminary results from the NRC analysis of these samples indicate agreement with the licensee data. The dose assessment from the preliminary dose rate survey indicates the dose to a member of the public from contaminated soil is approximately 1 millirem per year. The highest dose rate from licensed material found off-site was less than 2 millirem per hour on contact, although the dose rate decreased substantially at a distance of 10 cm. As material was located, during the licensee's initial scoping surveys, the locations were remediated to less than 0.5 millirem per hour and 10 millirem per year. Final remediation criteria have not yet been established.

Conclusions

The scope and depth of the licensee's current effort to review past radiological occurrences and assess significance are appropriate and sufficiently comprehensive for the site radiological characterization, as required by 10 CFR 50.82(a)(9)(ii). Areas have been identified, both on and off the site, that have measurable radioactive contamination that may require remediation. However, the maximum dose to an individual, including members of the public, from this contaminated material, in the locations examined to date, is below the regulatory limits in 10 CFR 20. There is a potential that the location or use of some of the material may have resulted in higher doses in the past. The final determination of this matter will require additional assessment by the licensee.

There were no records of surveys for excavation of soil outside the RCA. The licensee has recently sampled, analyzed and reported the results of contamination in the areas beyond the radiologically controlled areas. The radioactive material was not quantified and evaluated prior to being released in order to determine if it represented a significant pathway that should have been controlled and monitored in accordance with NRC regulations. However, any residual contamination on the site will be identified during the licensee's current site characterization program and will be evaluated for compliance with the decommissioning regulations for license termination.

The licensee is currently performing a full-scope radiological characterization of the site in order to safely decommission the facility. Continuing NRC inspections will monitor the licensee's regulatory compliance with the regulations.

3. LICENSEE MATERIAL RELEASE PROCESS

Scope

A selected review was performed of the licensee's procedures for conducting radiation surveys of materials to be released for unconditional use. The procedures required surveys and/or evaluations in accordance with 10 CFR Part 20 to ensure that licensed radioactive material was not inappropriately released. The review compared the instructions in the licensee's procedures against the guidance contained in NRC Information Notices, regarding what constituted a reasonable survey/evaluation.

Details

The earliest procedures available to the team were Standardized Procedure #17, Unconditional Radiological Release of Material Offsite, Revision 0, dated October 20, 1981, RAP 6.2-14, Unconditional Radiological Release of Material Offsite, Revision 0, dated January 28, 1982, and RPM 2.2-8, Unconditional Release Surveys, Revision 0, dated January 13, 1989. These procedures described the means by which material, that could potentially be contaminated, must be surveyed prior to being unconditionally released from the radiation controlled area.

The 1981 procedure did not contain release criteria guidance written in a practical format for use by the technician performing the survey. It appears that the procedure was quickly (within three months) superseded by RPM 2.2-8. The procedures from 1982 and 1989 provide specific instructions for the radiation survey of solid materials that may have fixed and/or removable surface contamination. The procedures specify that material containing detectable radioactive material, defined as 100 counts per minute above background, for beta-gamma surveys and 4 counts per minute above background for alpha particle surveys, is not to be released for unconditional release. The procedural guidance is consistent with early 1980s industry practices and NRC guidance published in Information Notice 81-07, "Control of Radioactive Contaminated Material (5/81)."

The information notice discussed that licensees are to perform adequate radiation surveys of waste with the potential to be contaminated with licensed material to ensure that licensed radioactive material is not inadvertently released. However, the notice specifically recognized that there would be levels of licensed radioactive material that could not be detected with commonly used radiation detection instruments and would be released into the general environment. The notice provided guidance on the minimum acceptable radiation detection capabilities for commonly used survey equipment; but, it did not provide release limits for radioactive material. The notice also acknowledged that there are other more sensitive analytical capabilities available to distinguish very low levels of radioactive contamination, noting that those capabilities are very elaborate, costly and time-consuming, making their use impractical (and unnecessary) for routine operations. Further, the notice stated that, based on the specified minimum detection capability, the potential radiation dose to members of the public from the release of any undetected, uncontrolled contamination would be significantly less than 5 mrem per year. This was considered by the NRC to be an acceptable dose criterion in 1981, since it was well below the explicit public dose limit of 500 mrem in 10 CFR Part 20. The industry generally viewed this information on required minimum detection capability for surveys as release limits. The NRC viewed licensee procedures that used the guidance in Information Notice 81-07 as acceptable.

In 1985, the NRC updated its radiological survey guidance for the unconditional release of potentially contaminated material to reflect the growing concern about the inadvertent release of licensed radioactive material. The updated guidance, which addressed the need for licensees to perform more sensitive surveys for large surfaces and packages of aggregated wastes, was published in Information Notice 85-92, "Surveys of Wastes Before Disposal from Nuclear Reactor Facilities (12/85)." The licensee's procedure, which was written in 1987 and referenced IN 85-92, did not address the updated NRC guidance. The absence of this updated and improved survey guidance in the licensee's procedure is not indicative of a good survey program for detecting surface contamination but not contrary to 10 CFR Part 20.

With respect to surveys for volumetric materials, NRC did not provide survey guidance or establish a release criteria for residual contamination. However, the licensee's procedure from 1982 had used an acceptance criteria for release if both of the following were shown by isotopic analysis:

> "Each isotope present does not exceed the exempt concentration specified in 10 CFR 30.70 Schedule A and that the sum of the isotope fractions is equal to or less than unity, and
>
> The total amount of each isotope present is not greater than the exempt quantity specified in [10 CFR 30.71] Schedule B."

Further, licensee records of surveys performed in the early 1980s denoted the inappropriate use of the values in 10 CFR Part 30, Schedule A, exemption tables. It appears that the licensee used these values as release limits, which is contrary to the requirements of 10 CFR Part 20. The dose impact from using the concentrations above the effluent release limits has not been evaluated because of insufficient information. The team believes that radioactive material in the public domain that has already been identified must be assessed by the licensee for the potential dose to the public.

For the radiation survey of a liquid or granular solid that may contain licensed radioactive material, the licensee's 1987 and 1989 procedures required that a representative sample of the material be counted on a gamma-ray spectrometry system. The system's lower limit of detection (LLD) for radioactive material is reported to be 3E-6 μCi/ml. The procedure states that this LLD corresponds to the most restrictive radioactivity concentration value in Appendix B, Table II, Column 2 of 10 CFR Part 20. It is further stated that, in practice, the system will be able to achieve a lower LLD than 3E-6 μCi/ml. Survey records from a December 1985 period that documented the survey of dirt in the RCA, using hand-held instruments, indicated that the licensee used a release limit of 1000 dpm/100 cm^2 and gamma analysis on a limited number of soil samples. Survey grid plans for the sampling size were not evident. Positive identification of licensed radioactive material was not acknowledged if the reported survey value was less than the isotope's concentration value in Schedule A, of 10 CFR Part 30 - exempt concentrations. For Co-60, a measurement result below 5E-4 μCi/ml[2] was apparently considered exempt based on the December 1985 survey record. This practice established release limits for radioactive material contained in solids intended for release to unrestricted areas and is contrary to 10 CFR Part 20, which only allows licensed radioactive material to be disposed of in specifically described ways.

Licensed radioactive material can only be disposed of in accordance with the methods described in 10 CFR Part 20. All other material that may be potentially contaminated with licensed material must receive a radiation survey. If any licensed radioactive material is detected, the material must be handled in accordance with 10 CFR Part 20. For the types of material and radionuclides typically observed at nuclear power plants, there are no release limits for detectable quantities of radioactive material contained in solid form released to unrestricted areas. Notwithstanding, the application of these procedures, containing inappropriate guidance, permitted the licensee to release solid materials that may have contained detectable quantities of radioactive materials, contrary to the requirements of 10 CFR 20.

For the radiation survey of a liquid or granular solid, the licensee's use of an LLD of 3E-6 μCi/ml and the exemption schedules from 10 CFR Part 30 were not consistent with NRC guidance published in Information Notice 88-22, "Disposal of Sludge from On-site Sewage

Treatment Facilities at Nuclear Power Stations (5/88)," and is contrary to 10 CFR Part 20. This Information Notice discussed the need for licensees to perform radiation surveys of representative samples of material under conditions that provide an LLD appropriate to measurements of environmental samples. Such measurements make it possible to distinguish licensed radioactive material from natural and fallout radioactive material. For the analysis of Co-60, the appropriate LLD for environmental samples of dry sediment is 1.5E-7 μCi/ml and 1.5E-8 μCi/ml for water samples. Thus, the licensee's survey program for liquids and solids was not able to adequately detect small quantities of licensed radioactive material within bulk quantities of liquid or granular solid material being released for unrestricted use. The licensee's stated LLDs were generally acceptable for use during the early 1980s, because the available gamma spectrometry systems at that time were not able to routinely achieve the low LLDs. Only expensive state-of-the-art systems could achieve the environmental LLDs. The NRC did not require licensees to have such sophisticated systems at that time. Consequently, licensees typically sent their environmental samples to a contractor laboratory for analysis, while surveys of bulk material for unrestricted release used less sensitive LLDs. However, since the early to mid-1990s, gamma-ray spectrometry systems that readily achieve the low LLDs are readily available at a reasonable cost. These systems are now routinely being used in the majority of nuclear power plants for routine use (i.e., release surveys of material).

The team reviewed the licensee's current procedures for the survey and release of material, RPM 2.6-16, Revision 7, dated 10/22/97 and RPM 2.2-22, Revision 0, dated 8/21/97. These procedures contained updated survey guidance which used more sophisticated equipment and techniques. For liquids and granular solids, the licensee's procedure required that the radiation surveys be performed to LLDs that are consistent with the environmental monitoring program. The licensee's procedures are generally consistent with current industry practices, NRC guidance, and 10 CFR Part 20. Other than using the dates on the licensee's procedures, the inspectors were not able to determine when the licensee updated survey criteria to use the environmental LLDs. Based on the guidance in RPM 2.2-8, the licensee was able to count lower than the stated LLD of 3E-6 μCi/ml. Thus, it appears that the licensee may have made a gradual transition to the use of the environmental LLDs over the years as new radiation detection equipment was installed.

<u>Conclusions</u>

Through 1989, the licensee's material release process for removable and surface contamination was generally consistent with NRC criteria and industry practices. However, it did not contain appropriate criteria for surveys of volumetric materials (i.e., soil, sludge and debris). Additionally, the licensee did not keep up with improvements within the industry to increase the sensitivity of radiation surveys. This deficiency was observed in the licensee's procedures, which had not incorporated updated NRC guidance for survey and release criteria from 1985 through 1989. The licensee's use of the exemption schedules from 10 CFR Part 30, Schedule A, in its survey and release procedures as a release criteria was not appropriate and contrary to the requirements of 10 CFR Part 20. Further, the use of the annual liquid effluent release concentration contained in Appendix B to 10 CFR Part 20 was also contrary to the requirements of 10 CFR Part 20. As a result, the licensee's survey procedures were not adequate to prevent the release of licensed radioactive material from the site and is contrary to 10 CFR Part 20. The licensee records show that radiation surveys were generally performed on most of the material released from the radiological control area of the site, in accordance with the procedures. Use of the inappropriate criteria resulted in radioactive material being inadvertently released from the controlled areas at concentrations above effluent release limits.

Dose assessments from prior release of material with residual contamination have not been completed. Dose estimates from recently identified materials in the public domain are well within the NRC annual exposure limits specified in 10 CFR Part 20. The team recognizes that the breakdown of the radiation protection program in 1975, which caused the release of the concrete blocks without an appropriate survey, could have resulted in exposures to the public in excess of 10 CFR Part 20 limits. However, preliminary assessments of the as found use and condition of the blocks (e.g., walkways, garden borders, foundation supports) have shown potential dose impact to the public to be negligible, to date. The licensee's current procedures for the survey and release of materials are consistent with current NRC guidance and 10 CFR Part 20.

4. RADIOLOGICAL ENVIRONMENTAL MONITORING REPORTS

Scope

A selected review was performed of the licensee's Annual Radiological Environmental Operating Reports to determine if licensed radioactive material of plant origin was observed in the environment outside the plant site. The licensee's reports were also reviewed for compliance with its radiological environmental monitoring program.

Details

NRC regulations require licensees to keep levels of radioactive material in effluents ALARA (as specified in 10 CFR 50.34a) to ensure that radiation doses to the public resulting from effluent releases or other radioactive material of plant origin will continue to remain minimal. To verify whether exposures in the environment are within the limits of 10 CFR Part 20 and to ensure that there is no long-term buildup of specific radionuclides in the environment, NRC requires licensees to monitor the environment for radioactivity of plant origin. This requirement is contained in General Design Criterion 64, "Monitoring Radioactivity Releases," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities." The licensee's radiological environmental monitoring program is designed around the NRC's requirements to establish correlations between levels of radiation and radioactivity in the environment and radioactive releases from the plant, as well as, to provide supporting evidence that the impact on the environment from plant operation is within the analysis contained in the plant's licensing basis documents (i.e., the Final Environmental Statement).

The review compared NRC regulations and regulatory guidance against selected examples of the licensee's Annual Radiological Environmental Operating Reports from 1979 to 1996. The licensee's reports contained adequate documentation of the required sampling, analyses, interpretations and discussion of results, historical trends, land use census, Quality Assurance program and a discussion of calculated dose commitments to a member of the public. The reports contained a discussion of the calculated radiation dose to a member of the public based on two methods: a calculation based on monitored radioactive effluents released into the environment and on calculations based on the concentration of licensed radioactive material observed in environmental media (fish, milk, vegetation, etc). The licensee reported that measurable levels of radioactive material, attributed to plant operation, were observed in selected environmental media. With the exception of tritium, all the reported concentrations of observed licensed material were within regulatory requirements and did not require a special report. As noted by the licensee, significantly higher levels of tritium than background were detected and reported. However, the calculated dose consequences to a member of the public from the radioactive material were within the regulatory requirements of 10 CFR 20 and Appendix I to 10 CFR 50.

The licensee's Annual Radiological Environmental Operating Reports discussed that elevated tritium concentrations were routinely observed in water samples obtained from the on-site indicator well. The licensee's reports noted the elevated levels and acknowledged that the tritium was a product of plant operation. The licensee further explained that the tritium in this well water was within an area influenced by radioactive effluents released in the discharge canal and that tritium has the capability to readily follow the flow of groundwater. The flow of this ground water is to the Connecticut river. The licensee states that the tritium in the groundwater and the river water has no dose consequence on the public, or plant workers, since the water is not used for drinking. Tritium was also observed above background levels in samples from the Connecticut River, but, at concentrations significantly lower than the samples from the on-site well.

The information contained in the licensee's reports was consistent with the guidance contained in Regulatory Guides 4.2 and 4.8 and Criterion 64 of Appendix A of 10 CFR Part 50. This regulatory guidance has remained essentially unchanged since it was introduced in the early 1970s.

In addition to the licensee's environmental monitoring program, the State of Connecticut performed independent environmental monitoring around the site. The state's program was partially funded by the NRC. The purpose of the state program is to obtain environmental monitoring data that is independent of the licensee's data. The state collects samples of environmental media from the same locations as the licensee and independently analyzes the samples. The results are reported to the NRC in an annual report. The review team examined the reports that were readily available for the years 1994 through 1996. The state reported "substantial agreement" between their data and the licensee's. No unusual conditions or levels of radioactive material were noted. The team noted that as of 1998, the NRC no longer provides funding to the states for independent environmental monitoring.

Conclusions

Overall, the radiological environmental monitoring data contained in the licensee's reports were developed in accordance with regulatory guidance, were properly documented, and were reported in accordance with Technical Specifications and regulatory requirements. No errors or omissions were identified. The licensee's radiological environmental monitoring program adequately established correlations between levels of radiation and radioactivity in the environment and radioactive releases from plant operation. It provided supporting evidence that the impact on the environment from plant operation is within the analysis contained in the plant's licensing basis documents (i.e., the Final Environmental Statement) and 10 CFR Part 20. No significant adverse environmental impacts were observed by the licensee's environmental monitoring program as a result of routine effluent discharges or from radioactive contamination that originated from the plant's RCA. The licensee's documentation is consistent with the findings of the Final Environmental Statement issued by the Atomic Energy Commission in October 1973. The review team did not identify any areas of the licensee's program, beyond those already identified in NRC inspection reports that were in violation of NRC regulations.

5. RADIOACTIVE WASTE SYSTEM

5.1 Licensing Design Basis

Scope

The review consisted of examination of documents from the issuance of the 1966 Haddam Neck Facility Description and Safety Analysis (FDSA) to the 1987 issuance of the Updated Final Safety Analysis (UFSAR). Documents reviewed included pre-operational safety analyses, the Provisional Operating License and amendments, the Full Term Operating License and amendments, the Final Environmental Statement, Facility Description and Safety Analysis (FDSA), Updated Final Safety Analysis Report (UFSAR) and plant design change requests related to radwaste systems. Documents were examined on site at Haddam Neck, in the NRC Region I offices and at NRC headquarters.

Background

The licensing basis of Haddam Neck's radwaste processing systems was examined to determine whether the location and use of the major systems were within the licensing basis. Other issues addressed included the extent to which spills of radioactive materials may be within the licensing basis and the interaction of fuel cladding defects on the design basis of radwaste processing systems. The adequacy of the installation of an extensive modification of the radwaste processing systems completed in 1975 was considered.

The licensing basis includes NRC regulations, the license, orders, exemptions, technical specifications, the Final Safety Analysis Report (FSAR) and plant modifications, among other documents. Because the regulations, licensing documents and the plant itself changed over time, the licensing basis also changed. In addition, the understanding of what constitutes the licensing basis has changed over time by widening what was included in the definition. The overall effect of the changes has been to increase the margin of safety associated with nuclear power plant operation and to provide greater quality assurance through more extensive documentation.

The licensing basis defines the design and operation of a nuclear facility to provide several layers of defense-in-depth protection of the public health and safety. The health effects of radiation have been well studied. Accordingly, regulatory limits are established well below levels that cause harm, so that operation of a nuclear power plant within regulatory limits will cause no significant public health and safety effects. To assure that regulatory limits are not exceeded, the licensing basis adds a margin of safety by establishing safety limits that are more conservative than the regulatory limits. The safety limits include surveillance requirements so that the licensee will observe the condition of the plant and take corrective action in a timely manner. Sound design and high quality established by the licensing basis minimizes the possibility that malfunctions can occur. However, the plant design includes provisions, such as requiring multiple systems to perform important functions, to safely contain radioactive materials even if some equipment does malfunction or if a mistake is made. If multiple system failures should nevertheless occur, emergency procedures provide methods to mitigate the consequences of an accident and protect the public.

The defense-in-depth philosophy has been successful in preventing any significant public health and safety effects due to the operation of nuclear power plants in the United States.

Details

Connecticut Yankee Atomic Power Company filed its Facility Description and Safety Analysis (FDSA) for the Haddam Neck nuclear plant on July 19, 1966. Although not specifically mentioned, it is clear from the descriptions and knowledge of the plant layout that some waste handling would necessarily have to occur outside of buildings and enclosures.

The FDSA notes that the design basis of the radwaste systems included the assumption that 1% of the fuel fission products would be released into the reactor coolant by diffusion out of the fuel pellets and through cladding defects. The gaseous waste treatment system design used a somewhat different criterion, by addressing the magnitude of potential releases due to defects in 1% of the fuel rods.

Tritium was addressed in the FDSA by assuming 50% of its production would be released into the reactor coolant. Calculations demonstrated that if all the expected tritium inventory in the reactor coolant system (4015 Ci/yr) was released to the environment, the average concentration in effluents would still meet 10 CFR 20 limits.

The U.S. Atomic Energy Commission (AEC) issued a safety evaluation in connection with the plant's proposed Provisional Operating License on May 12, 1967. AEC accepted the design basis of defects in 1% of the fuel. The staff concluded that normal operation within the limits of the technical specifications would not result in potential offsite exposures in excess of 10 CFR 20 limits. The original technical specifications did not contain a fuel cladding defect limit.

The 1967 safety evaluation noted that the storage and hold-up facilities were located "outside containment" and analyzed the consequences of a waste gas sphere rupture. The dose at the site boundary was found to be within 10 CFR 20 limits. It was also noted that the Connecticut River was not used for drinking water supplies downstream from the site. Therefore, an accidental release of radioactive liquids into the river from the plant would not affect public water supplies.

Haddam Neck received its Provisional Operating License on June 30, 1967, which authorized operation at approximately 80% power. The plant began commercial operation on January 1, 1968. Full power operation was authorized on March 3, 1969.

AEC issued a safety evaluation on July 1, 1971, in response to the licensee's request for a full-term operating license. The SER noted that subsequent to the issuance of Haddam Neck's Provisional Operating License, the Commission had published General Design Criteria (GDC), effective May 21, 1971. The staff found that Haddam Neck conformed to the intent of the GDC. A design change to the radwaste systems was noted. The change allowed use of demineralizers in place of the originally installed aerated liquid waste evaporator, which had not met performance expectations, and accommodated liquid waste flow rates that exceeded the original design values. The staff concluded the design change met the ALARA criteria. The overall conclusion was that continued operation of the Haddam Neck plant would not endanger the public health and safety.

By 1972, the licensee was aware that its existing radwaste systems would not meet the requirements of proposed 10 CFR Part 50, Appendix I. An extensive modification was initiated in 1972, and made operational in 1975, to meet the new limits. The modification relocated the waste gas sphere from an outside area to a newly built waste-processing building. In addition, an evaporator was added to process waste liquids.

In December 1972, the licensee sent the design of the radwaste system modification to AEC. The licensee committed to issue an amendment to its license application upon completion of the modification. AEC acknowledged receipt of the design. However, no record can be found to demonstrate that the licensee submitted a license application amendment. The FSAR, reissued on October 15, 1975, and last updated in 1981, described only the original plant radwaste treatment equipment, not the modified system which had been operating since July 1, 1975.[1] A description of the modification was later included in the 1987 issuance of the UFSAR.

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1. Haddam Neck was not required by regulation to make periodic updates to its FSAR prior to 1987. NRC considered the need to require periodic updating of the Final Safety Analysis Report in proposed rulemaking published for comment on November 8, 1976 (41 FR 49123). At the time, there was no requirement for a licensee to incorporate revisions, changes or amendments into the FSAR except where a hearing was held on an operating license application. After Haddam Neck received its FTOL in 1974, no updates to the FSAR were required until the FSAR updating rule became effective July 22, 1980 (45 FR 30614). However, Haddam Neck was exempted from the rule due to its participation in the SEP.

NRC announced the Systematic Evaluation Program (SEP) for a number of older plants in November 1977. The objective was to determine and document the degree to which they met licensing requirements for new plants. Haddam Neck was among those affected. As a result, Haddam Neck was exempted from the requirements of 10 CFR 50.71(e) to update its FSAR until after the SEP was complete. A number of extensions to the completion date was issued such that the licensee was not required to submit an updated FSAR until June 30, 1987. Haddam Neck made the submittal on June 22, 1987.

AEC completed the Final Environmental Statement for Haddam Neck in October 1973. The environmental impact of the plant as it existed at the time was evaluated and found not to endanger the public health and safety. Among other items, the FES considered the release of tritium to the environment. AEC estimated that all tritium produced in the core (8000 Ci/yr) could be released without exceeding regulatory limits.

AEC evaluated the expected performance of proposed radwaste systems modifications on normal effluents in the October 1973 Final Environmental Statement (FES). The FES contained simplified diagrams of the anticipated changes. The FES concluded that dose to individuals was within design objectives and ALARA. Dose to the population in the 50-mile radius was a small increment of natural background fluctuation, considered to be immeasurable and constituting no meaningful risk. The calculated population dose was lower for the modified radwaste system design than the original design.

The October 1973 FES considered the radiological impacts of a series of postulated accidents using the proposed guidance of 10 CFR 50 Appendix D, Implementation of NEPA. (Appendix D was revoked when 10 CFR Part 51 incorporated NEPA requirements.) The basis of the evaluation was the original plant design. Included in the consideration were Category 2 accidents, accidental spills and releases of radioactive materials outside containment, including those due to such developments as relief valve actuation. Doses were found to be within 10 CFR 20 limits for this category. Accidents analyzed in Category 3, Radwaste System Failures, included analysis of a rupture of the waste gas decay sphere specified in the original plant design. The consequences were within 10 CFR 20 limits, assuming an operable radiation monitoring system and that the licensee took some mitigating actions. Whole body dose for that accident was calculated as 0.185 rem at site boundary. Category 5 accidents involved release of fission products to primary and secondary systems due to fuel cladding defects, primary to secondary leakage and steam generator tube rupture. The consequences were within 10 CFR 20. Table A (adapted from Table 7.2 Radiological Consequences of Postulated Accidents Final Environmental Statement, October 1973) indicates several of the doses calculated in the FES.

TABLE A

Class	Event	Dose at Site Boundary
1	Trivial incidents	Within 10 CFR 20[1]
2	Small release outside containment	Within 10 CFR 20[1]
3	Radwaste equipment leakage or malfunction	0.046 rem
3	Release of waste gas storage tank contents	0.185 rem
3	Release of liquid waste storage contents	0.005 rem
5	Fuel cladding defects and steam generator leaks[2]	Within 10 CFR 20[1]
5	Off-design transients that induce 0.5% fuel failure and steam generator leak[2]	.001 rem

1. The applicable standard was 0.5 rem whole body or equivalent dose to an organ. Where no specific dose value is listed, releases were expected to be a small fraction of 10 CFR 20 limits for liquid or gaseous effluents.
2. Leakage other than a tube rupture, which was analyzed separately.

Thus, the FES anticipated occasional spills, lifting relief valves on radwaste holdup tanks and fuel cladding defects in the assessment of dose consequences. The FES found that these operational occurrences would not endanger the health and safety of the public because the potential off-site doses were below regulatory limits.

In August 1974, an inspection of the existing radwaste systems found them in compliance with the FSAR and Technical Specifications. The inspection was done prior to the operation of the modified radwaste systems.

A Full-Term Operating License (FTOL) was issued on December 27, 1974. A supplement to the safety evaluation issued with the FTOL noted that radioactive releases for 1970 through 1973 were well within the limits of the plant Technical Specifications. It further noted that augmented effluent treatment systems were expected to be in operation in 1974, which would produce significant improvement in releases, meeting ALARA guidelines. That conclusion was conditional upon the licensee properly operating and maintaining the equipment. The supplemental SER further concluded that the Haddam Neck facility was in conformance with all rules and regulations of the Commission.

The modified radwaste systems were put in operation on July 1, 1975. The NRC found the design acceptable. However, during construction, field changes were made to substitute rupture discs for safety valves on several tanks, such as the waste gas decay tanks and degasifier. The changes were made due to the long lead time to deliver safety valves. No safety evaluation was done for the change by the licensee as required by 10 CFR 50.59, which had been in effect since 1969. The discs ruptured on several occasions before the licensee, with recommendation from NRC, replaced them with safety valves in 1981.

Haddam Neck submitted its final Demonstration of Compliance with 10 CFR 50, Appendix I, on November 1, 1976. The report noted average annual tritium releases were 5761 Ci/yr. It also noted that "uncontaminated drains" were expected to contain liquid with activity about 1% of primary coolant activity. The liquids would be treated prior to release.

An internal NRC memo dated October 14, 1977, contains a detailed evaluation of the radioactive waste systems at Haddam Neck. It concludes that the modified systems were capable of maintaining releases ALARA and within the levels required by Appendix I.

Haddam Neck experienced an unplanned noble gas release in excess of Technical Specification concentration and release rate limits on December 16, 1979. A rupture disc on the degasifier (one of the modified radwaste system components) actuated due to overfilling with water. The overfilling occurred due to failure of the level control relay to stop flow. The dose at the site boundary was calculated at 0.00045 rem. (Comparing this value to the Table A-1 event, "Radwaste equipment leakage or malfunction," it will be seen that the off-site consequences were within the bounds of the FES.) The root cause was attributed to design errors in that a rupture disc was used for pressure relief rather than a safety valve, which would reset once pressure decreased. The root cause analysis did not recognize that the original design specified safety valves, and that rupture discs were substituted during construction.

The licensee considered several actions in response to the 1979 gas release. Two were implemented. The first, PDCR 345, added a liquid level alarm to alert operators that water was collecting in the base of the plant stack. The change was initiated in January 1980 and received its final QA review on September 17, 1982. The second documented action taken was replacing the rupture disc with a safety valve. This was initiated on September 18, 1981, as PDCR No. 413, and given final QA review on September 13, 1982. The design document notes that Haddam Neck took the corrective action in response to an NRC requirement. The requirement was incorporated as an addition to the requirements of NUREG-0578 (Systems Integrity). The design document notes that a total of five unplanned releases in the previous four years had occurred due to rupture disc actuation. Rupture discs were used on the waste gas decay tanks and steam generator blowdown tanks, as well as the degasifier. Some of the discs, not specifically identified, were noted as relieving directly into the PAB or Waste Disposal Building. The building ventilation systems discharged to the plant vent stack, which was a monitored release path. Subsequently, all the rupture discs were replaced.

Radiological Effluent Technical Specifications (RETS) were incorporated in the Haddam Neck Operating License on September 5, 1985, as License Amendment No. 68. The safety evaluation noted the purpose of the proposed technical specifications was to keep releases to the environment ALARA during normal operations and expected operational occurrences. The technical evaluation of the licensee's proposal was done by a contractor whose report is incorporated into the SER. Section 3.1.1 of that report states that "Liquid radioactive wastes are collected in sumps and drains in the various buildings, then transferred to the appropriate tanks in the radwaste building for further treatment." Relative to this description, the NRC team noted that an unplanned, unmonitored liquid release occurred in 1989 when workers processed several drums of containment sump water in the spent fuel building. The processing done in the spent fuel building appears to have been outside the licensing basis of the RETS. The workers treated the water by filtering, and directed the filtrate to the floor drain under the spent fuel pit heat exchanger. The workers believed the drains went to the radwaste system. In fact, the drains led to the yard drains, which allowed the water to leave the RCA via an unmonitored path (See Appendix B for more details).

In 1989, Haddam Neck found 456 fuel pins with throughwall cladding defects during the refueling outage. The defects were caused by machining chips left in the core after thermal shield modifications done during the previous refueling outage. The NRC team noted that the waste gas decay tank accident analysis as described in both the FDSA and UFSAR assumes 1% (320 rods) fuel cladding defects as a design basis. Because the defects observed in 1989 released relatively small amounts of iodine into the reactor coolant system during normal operation, the licensee's fuel monitoring program anticipated only 10 to 12 failed rods prior to refueling. Although the event was reported to the NRC, the licensee did not recognize that the number of defected rods exceeded an accident analysis design basis when the extent of the damage was determined after plant shutdown. However, the actual curie content of the tanks did not exceed 5% of the activity assumed in the accident analysis for purposes of calculating off-site dose consequences.

Conclusions

The original design and safety evaluations anticipated radwaste handling outdoors. As of 1974, the radwaste systems were in compliance with the FDSA and technical specifications. Operational occurrences resulting in spills and releases outside containment were evaluated in the Final Environmental Statement and all were found to be within the regulatory limits for protection of the public.

The design of modifications to the radwaste processing system completed in 1975 met Appendix I requirements. During construction, a field change was made to the design to substitute rupture discs for safety valves to provide pressure relief protection on several tanks. The licensee did not perform a safety evaluation of the change, as required by 10 CFR 50.59. In addition, the field change appears to have met the definition of an unreviewed safety question (USQ), in that a malfunction of a different type than previously evaluated may have been created. If the change was a USQ, prior NRC approval would have been required to make the change. The rupture discs were replaced with safety valves after an unplanned release that occurred in 1979.

Liquid waste processing in the Spent Fuel Building resulted in an unplanned release in 1989. The processing did not conform to the conditions analyzed in the Safety Evaluation Report performed for the 1985 RETS license amendment.

The waste gas decay tank rupture accident analysis assumed 1% (320 rods) defected fuel as a design basis. In 1989, Haddam Neck found 456 defected rods during the Cycle 15 refueling outage. Operation during Cycle 15 appears to have been, in part, outside the design basis for that accident. The fuel monitoring methods used during operation underestimated the number of leaking rods due to the small amount of iodine produced by the defects. After inspecting the fuel and discovering the full extent of the cladding defects, the licensee did not recognize that the 1% design basis for fuel integrity had been exceeded. However, the amount of radioactive gas in this waste gas decay tank was well below the design value used to calculate off-site dose consequences.

5.2 System Operations

Scope

This section reviewed the licensee's procedures and program for the transfer of liquid radioactive material in radioactive waste systems.

Details

The licensee's liquid radioactive waste-handling facilities required transfer of radioactive slurries outside the confines of plant buildings. This practice was not uncommon among nuclear plants licensed in the 1960s, such as Haddam Neck. Haddam Neck's design called for resin liners to be contained in designated pits providing shielding for personnel and dikes for containment of potential spills. Resin liners were stored outside in unroofed areas until 1981, when a spent resin storage facility was built.

The Process Control Program (PCP) for Haddam Neck was proposed in 1979 by the licensee and described the functions of the Liquid Waste System and Purification System. The purpose of the PCP was to ensure that the radioactive waste liquid solidification system was operated to produce a final product that contained no free-standing water and resulted in a completely solidified waste. A PCP is required to ensure that waste is properly characterized as required by 10 CFR 61.56. Liquid radioactive waste that required solidification was processed as directed by approved procedures. The PCP also described the purification system functions, which were to remove impurities from the reactor coolant system during operation or plant shutdown, the volume control tank and RWST, the reactor cavity during refueling and the spent fuel pit water, when necessary. The PCP provided for sluicing of resins to a shipping container in a reinforced concrete shipping cask using demineralized water, which was pumped back to the aerated drain tanks for further processing. The proposed PCP contained details of the process by which concrete was added to radioactive wastes in certain prescribed ratios to form an acceptable waste form for disposal.

A revised Process Control Program for Haddam Neck became effective in 1985 with Amendment 68 incorporating the Radiological Effluent Technical Specifications (RETS) to Appendix A of the operating license. These changes followed the implementation of changes to 10 CFR 20 regarding low-level radioactive wastes and the incorporation of the new 10 CFR 61. The PCP states that Haddam Neck is committed to a management system and procedures necessary to ensure that:

1. all liquid wastes are solidified in accordance with regulatory and disposal site criteria;
2. containers, shipping casks and methods of packaging meet 10 CFR 71 and 49 CFR requirements;
3. waste classification will meet 10 CFR 61 and disposal site requirements, and;
4. approved procedures will include detailed information regarding sample mixing, solidification processes, QA of the solidification process, absence of free liquids, and handling containers if solidification is exothermic.

The stated objective of the Haddam Neck Process Control Program was to ensure safe, effective solidification of radioactive waste liquids and slurries for off-site disposal and to ensure compliance with 10 CFR 71, 10 CFR 61, 10 CFR 20, 49 CFR and disposal site regulations. The details required to meet these commitments were maintained in approved procedures. In 1986, an expanded facility for low-level radioactive waste (LLRW) handling and storage was built.

Subsequent inspections reviewed radioactive waste handling practices and the licensee's PCP. One inspection in 1986 identified four weaknesses in the classification of wastes for Iron-55 and licensee internal audits of the PCP. Corrective actions were implemented and closed out during an inspection in 1987. During an inspection in 1991, the NRC reviewed changes to the radwaste system, including the installation of a spent resin storage tank. The report noted that spent resins were primarily processed by dewatering using a vendor-supplied system. The Process Control Program for both methodologies (e.g. dewatering and cement solidification) were examined by the inspector and determined to be appropriate.

Available radwaste operations procedures controlling the transfer of radioactive slurries to shipping containers revealed that the licensee continued to maintain procedural control over such transfers. The revisions that were reviewed included the following:

"Spent Resin Storage Facility, RPM 3.3-1, Rev.3", 9/19/94
"Set-up of HICs for Resin Slurry, RPM 3.4-2, Rev.4", 5/14/93
"Dewatering of HICs in the Spent Resin Storage Facility,
RPM 3.4-4, Rev.14", 11/22/96
"Spent Resin System Operation, RPM 3.4-6, Rev.8", 12/12/96
"Resin Slurry to Spent Resin System, RPM 3.4-8, Rev.3", 12/12/96
"Shipment of Radioactive Waste Packages, RPM 3.6-1, Rev. 9", 2/11/97
"Set-up and Test of the Chem-Nuclear Set-Up and Test of the Chem-Nuclear
Universal Dewatering System, RPM-3.9-8, Rev.3",(Major), 2/15/94

Copies of earlier procedures were not readily available, however, the above procedures included caution statements for control of contamination. The NRC team noted that though there were some incidents regarding radioactive drain transfers, resin spills, cask washdown and contamination from outside storage, these incidents were infrequent. Because these contaminating events occurred outdoors and the boundary of the RCA was close, contamination may have spread to adjacent areas.

Conclusion

Haddam Neck's controls for operating the liquid radioactive waste systems in outdoor locations within the radiological control area appeared adequate over the duration of commercial operation. The licensee maintained approved procedures for radioactive waste-handling operations in accordance with their license requirements. However, some of the outdoor practices may have resulted in the spread of contamination to areas on the licensee's property that were not included in the licensee's survey program. A Process Control Program describing the liquid radioactive waste and purification systems was maintained with appropriate procedural controls. When regulatory requirements changed, the PCP was revised accordingly. Although violations of specific requirements were identified early after the implementation of the revised PCP in 1986, the Process Control Program at Haddam Neck was found to be appropriate.

6. LICENSEE RESPONSE TO IE BULLETIN 80-10

IE BULLETIN NO. 80-10: CONTAMINATION OF NONRADIOACTIVE SYSTEM AND RESULTING POTENTIAL FOR UNMONITORED, UNCONTROLLED RELEASE OF RADIOACTIVITY TO ENVIRONMENT

Scope

This section reviewed the licensee's response and NRC inspection follow-up to actions required by NRC IE Bulletin 80-10. By the end of June 1980, licensees were required to: (1) review their facility design and operation to identify systems that are considered non-radioactive but could possibly become radioactive through interface with radioactive systems (i.e., become contaminated due to leakage, valving errors or other operating conditions); (2) establish a routine sampling or monitoring program for these systems to promptly identify any contaminating events which could lead to unmonitored, uncontrolled liquid or gaseous releases to the environment, including releases to on-site leaching fields or retention ponds; (3) restrict access to contaminated non-radioactive designed systems or evaluate operation in accordance with 10 CFR 50.59, and consider the level of contamination to the radioactive effluent limits of 10 CFR Part 20, RETS and to environmental radiation dose limits of 40 CFR 190; and lastly (4) determine if potential releases comply with requirements for radioactive effluent releases or, if continued operation required a change to technical specifications or constituted an unreviewed safety question, not operate the system as contaminated without prior NRC approval. The Bulletin also stated that if a nonradioactive system was contaminated, decontamination should be performed as soon as possible.

The licensee's original response to IE Bulletin 80-10 was evaluated and found adequate by the NRC staff, but an Inspector Follow-up Item (IFI) was opened due to the licensee's failure to address non-liquid systems in its response. The item was closed in January 1983 (IR 50-213/82-21) by the resident inspectors. The item was opened again in 1990 by a radiation specialist due to positive levels of I-131 detected in vegetation samples close to the site boundary. The IFI was closed again in 1991, when the licensee demonstrated the low safety significance of the non-liquid systems (as addressed in 1982) and the corrective actions to prevent future unmonitored releases. The NRC staff stated in 1991 that the licensee's response was adequate, however, several unmonitored spills and releases outside the radiation controlled area have occurred since the licensee evaluated Bulletin 80-10.

Details

The licensee prepared documentation of all relevant information to IE Bulletin 80-10, including their response. The various dates involved with the licensee's IE Bulletin 80-10 response are listed below:

May 6, 1980:

Issuance of IE Bulletin No. 80-10, titled "Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to Environment." The bulletin described a problem at the Brunswick Nuclear facility where the auxiliary boiler was operated for an extended period of time with radioactive water. A tube leak in the firebox of the boiler resulted in an unmonitored, uncontrolled release of radioactivity to the environment. Action items were to be taken by each licensee with a verification letter submitted to the Regional NRC Office.

June 23, 1980:

Licensee submitted response to IE Bulletin 80-10, EN-MO-153. The response stated that actions for Bulletin items 1 and 2 were completed.

February 8, 1982:

Region I radiation specialist reviewed the licensee's documentation of the review performed for IE Bulletin 80-10. The inspector found that the licensee's review did not include non-liquid systems. An inspector follow-up item was opened (IFI 81-11-01). The licensee committed to perform the non-liquid systems review by 11/30/82.

January 3, 1983:

NRC Inspection Report No. 50-213/82-21 documented the licensees actions for non-liquid systems in response to IE Bulletin 80-10. The review was documented in the licensee's report, CYAPCo CN 82-803, dated 11/29/82. The licensee concluded that there was a very low probability that contaminated releases could occur through the non-liquid systems. The follow-up item was closed.

May 17, 1989:

The cover letter for NRC Inspection Report No. 50-213/89-02 stated that the Regional staff was concerned regarding an unmonitored release path that had existed through the Spent Fuel Building floor drains and that the radioactive liquid entered these drains on at least one occasion. The letter stated that the issue of unmonitored release paths was brought to the licensee's attention in IE Bulletin 80-10 and the area warranted further consideration.

June 29, 1990:

NRC Inspection Report No. 50-213/90-11 documented a specialist inspector's review of the licensee's response to IE Bulletin 80-10 and the problem with the clean drain in the Spent Fuel Building that had been used to dispose of contaminated water. The inspector also noted that positive levels of iodine (I-131) found in vegetation samples close to the site boundary could be associated with an unmonitored release path. The inspector requested that the licensee complete the evaluation of non-liquid pathways to close the follow-up item (the inspector did not realize that the item had already been closed in 1983), verify that the remedial action for unmonitored pathways in the original and subsequent evaluations was complete, and review the adequacy of the original engineering evaluation of unmonitored pathways conducted in 1980 (EN-MO-153) in view of the environmental sampling results I-131 in vegetation. The inspector noted that this was an unresolved item (URI 50-213/90-11-01).

January 24, 1991:

NRC Inspection Report No. 50-213/91-01 documented a specialist inspector's follow-up of the URI 50-213/90-11-01. The inspector noted that the I-131 found in vegetation could be explained by previous known releases of noble gases and iodine that were higher than normal releases because of significant fuel cladding defects. The inspector also reviewed the unmonitored release from the Spent Fuel Building to the open trench surrounding the 115 kV switchyard. The inspector noted that the pathway had been identified in the licensee's original review for IE Bulletin 80-10. The licensee had stated the drain was plugged after the pathway was identified in 1980. Sometime between 1980 and 1989, the plug had been removed. As a new corrective action, the licensee plugged the drain line and welded the plug in place to prevent inadvertent removal. The licensee also revised the procedure for monitoring potential pathways to the environment. The revised procedure included plugging and labeling drains, as well as development of a surveillance program to ensure that the pathways are monitored at appropriate frequencies to ensure the systems remain noncontaminated. The inspector closed the item based on the licensee's corrective actions.

The licensee had another unmonitored, uncontrolled release from the Primary Auxiliary Building (PAB) heat exchanger through a drain line to an area drain for the Adams Filter dike in April 1994. The drain emptied into the open trench in the 115 kV switchyard. The total radioactivity released was not significant, but the event cause was attributed to the lack of controls for the drain systems in the radiologically controlled area.

In the period between 1996 and 1997, NRC inspectors questioned releases to the environment which prompted a new review of IE Bulletin 80-10 by the licensee's staff. Two separate contractors were reviewing the potential for non-contaminated systems to become contaminated and the historical information related to past contamination of clean systems. The findings appear to indicate that the initial response to the bulletin by the licensee's staff was a minimal system review and the licensee's program did not require safety evaluations when nonradioactive systems became contaminated. Consequently, several systems that became contaminated did not have a safety evaluation performed. Those systems included the closed cooling water system, the drain systems, the component cooling water system and the turbine building waste water system. For example, the main turbine was known to have contamination from steam generator sludge contaminated by primary to secondary leakage as early as 1970, yet a 10 CFR 50.59 safety evaluation had not been completed for the turbine sump. This was in direct conflict with the guidance in IE Bulletin 80-10. The contractors made recommendations that the licensee is currently reviewing.

The NRC Region I staff inspected the licensee's progress in reviewing the implementation of the guidance from IE Bulletin 80-10. This review is documented in NRC Region I Inspection Report No. 50-213/97-10. The licensee developed and implemented a three-phase program to re-evaluate plant systems relative to NRC Bulletin 80-10 criteria. The program included review of all systems, including current systems in operation and abandoned systems. The licensee performed a comprehensive review of the systems relative to criteria contained in NRC Bulletin 80-10 and was establishing a sampling program to monitor those systems, as appropriate, to ensure detection of potential cross-contamination of normally non-radioactive systems. The review was completed on November 14, 1997. The review of known radioactive systems was for purposes of evaluating system interfaces with typically non-radioactive systems. The review of non-radioactive systems included review of system interfaces and past known contamination history. The licensee developed a safety evaluation status summary for affected or potentially affected systems and was performing safety evaluations for systems considered high priority (i.e., systems known to contain or that had contained radioactive material or had a high potential for contamination.)

The licensee had also established a sampling and analysis matrix for use in evaluating proposed changes to the chemistry sampling program. The licensee revised analysis methods to establish lower limits of detection to meet environmental lower limits of detection. The licensee was also initiating action to review and revise the radiological environmental monitoring program and the off-site dose calculation manual to provide for sampling of alternate release paths (e.g, storm drain system) as appropriate. The potential changes to the off-site dose calculation methods included addition of the external containment sump and RCA yard drain system as a continuous release pathway.

The inspectors noted that although several systems were identified that exhibited low-level contamination (e.g., closed loop cooling, heating and condensate steam component cooling water, turbine sumps) no apparent immediate safety concerns were noted. The licensee had posted the turbine building with information signs indicating the need to contact radiation safety personnel when planning work in the turbine building on a potentially contaminated system.

<u>Conclusions</u>

A recent review by the licensee relative to performance on IEB 80-10 revealed that the initial review was not fully comprehensive because it did not identify all systems that could be potentially contaminated. The recent review also revealed that noncontaminated systems had been used after they were contaminated, and that no safety evaluation had been performed. Very recently, the licensee's implementation and evaluation for continued use of contaminated systems were reviewed by the NRC and was documented in NRC Region I Inspection Report No. 50-213/97-10. The inspectors found that the licensee had established and implemented a task force to reconsider the guidance of NRC Bulletin 80-10, and develop a comprehensive program for decommissioning. This was considered a very good initiative to improve management oversight of station systems that could become cross-contaminated and result in an unmonitored release to the environment.

7. PLANT EXPERIENCE WITH STAINLESS STEEL CLAD FUEL

Scope

The review summarized the licensing and performance history of stainless steel clad fuel at Haddam Neck. The purpose was to determine if the operation of the Haddam Neck plant was within the licensing basis of the fuel design and to identify areas to be considered during site characterization.

The review included examination of documents from the 1967 issuance of the safety evaluation for the Provisional Operating License to the 1994 issuance of License Amendment No. 171, which removed certain restrictions pertaining to stainless steel fuel because the licensee had switched to zircaloy clad fuel. Other reactor systems and components were not examined, except to note an ECCS reanalysis done in 1981 which affected fuel peak clad temperature.

Documents reviewed included pre-operational safety analyses, the Provisional Operating License and amendments, the Full Term Operating License and amendments, the Final Environmental Statement, Facility Description and Safety Analysis (FDSA), Updated Final Safety Analysis Report (UFSAR) and plant design change requests related to radwaste systems. Documents were examined on site at Haddam Neck, in the NRC Region I offices and at NRC headquarters.

A limited scope review was performed of the Haddam Neck power plant's reported releases of solid, liquid and gaseous radioactive material through monitored, unmonitored and uncontrolled pathways. The reviewers compared related NRC regulations and guidance to selected examples of the licensee's effluent and environmental reports from 1979 to 1996.

Background

The defense-in-depth approach establishes four major barriers to isolate fission products from the environment. The fuel is formed into hard, dense, ceramic pellets of uranium oxide which have the capacity to retain a large fraction of the fission products. This provides the first barrier to fission product release. The pellets are sealed inside a metal tube, which is the fuel cladding. A tube filled with pellets is a fuel rod. The cladding forms the second barrier. The fuel rods are arranged in square bundles, held together with several metal plates. The bundled rods are called fuel assemblies. These assemblies are placed in the reactor vessel for use in electricity production. The reactor vessel and associated equipment and piping used to circulate water through the fuel assemblies are the reactor coolant system. This system forms a third barrier to fission product release. The primary system is housed in a metal-lined concrete structure called a containment, which is the fourth major barrier.

As long as any one of the barriers outside the fuel pellets remains intact, fission product releases to the environment can be controlled to levels below regulatory limits for protection of the public. Regulations limit the amount of leakage allowed from each barrier. The leak rates may be directly specified, as with containment structures, or implicit, as in maintaining the fuel in a coolable geometry.

In addition, power plants use active systems to remove radioactive material from the reactor coolant system for processing. The systems collect radioactive materials in various tanks, which hold them for a period, allowing time for decay. Afterwards, the materials are released in a controlled manner or packaged and shipped for disposal.

The review that follows discusses fuel defects that formed throughwall penetrations of the cladding, no matter how small, that allowed fission products to enter the reactor coolant system.

Overview

Haddam Neck was one of the few commercial nuclear plants to use stainless steel fuel cladding. The NRC and its predecessor, AEC, analyzed the fuel performance on several occasions and found that the design met regulatory requirements. The Electric Power Research Institute (EPRI) examined the performance record of over 550,000 stainless steel clad fuel rods in the United States and Europe in a report published in 1982. EPRI concluded that stainless steel clad fuel performance was excellent overall, with the exception of certain specific cases, such as the Haddam Neck cladding defects in 1979.

In 1992, Haddam Neck began a conversion to zircaloy cladding to reduce fuel costs. The conversion was completed in 1995.

Haddam Neck experienced throughwall fuel cladding defects in the range of 45 to 456 rods in Cycles 8, 15, and 16, which occurred in 1979, 1989 and 1991, respectively. The Haddam Neck reactor vessel contained 157 fuel assemblies, with a total of 32,028 fuel rods. Fuel performance and licensee actions during those cycles were examined to determine the extent to which the licensee's action conformed with the licensing basis.

7.1 AEC/NRC Evaluations

In the first seven years of plant operation, the stainless steel clad fuel used at Haddam Neck received three major evaluations from AEC and NRC staff. A fourth major evaluation was performed by the NRC in 1983. This last evaluation was issued in License Amendment No. 52, as noted in Section 7.2, to approve a new fuel design. The first safety evaluation was published in May 1967 to support the Provisional Operating License. It was noted that the fuel design was similar to other operating plants using stainless steel.

The staff concluded that operation within the Technical Specifications would not cause cladding defects in excess of the design basis (1% of the fuel rods). The second evaluation, in July 1971, was done to support issuance of a Full-Term Operating License (FTOL). It noted that upgrades to the ECCS reduced the calculated peak clad temperature (PCT) by 50°F. A third evaluation was published in December 1974. The FTOL issuance had been delayed to prepare an environmental impact statement in accordance with National Environmental Protection Act. Due to the delay, the FTOL safety evaluation was updated. The staff noted that Haddam Neck fuel performance was bounded by conditions at San Onofre Unit 1, and concluded that the likelihood of clad collapse was remote. A re-analysis of peak cladding temperature by the licensee using an updated Westinghouse model calculated the PCT as 2300°F. The staff concluded the plant met the Interim Acceptance Criteria for ECCS.

7.2 Defects in Cycle 8, 1979

Elevated reactor coolant iodine levels at the end of Haddam Neck's Cycle 8 in 1979 indicated fuel cladding defects. The fuel inspection during refueling revealed 36 leaking assemblies, containing about 45 leaking rods. All the assemblies came from one batch, Batch 8, which used BNFL-supplied pellets and Babcock & Wilcox (B&W) fuel fabrication services. Batch 8 assemblies had the highest burnup (29,000 to 36,000 MWD/MTU) in the core. The faulted assemblies were removed from service. The fuel pellet supplier, British Nuclear Fuels Ltd. (BNFL) suggested that a power ramp at the end of Cycle 7 may have initiated the defects. Power ascension restrictions were put in place until the cause of the defects was determined.

For Cycle 9, six assemblies of Batch 9, which had seen service in Cycle 8, were loaded into the core. These assemblies had an average burnup of 24,200 MWD/MTU prior to loading. They contained 162 rods with pellets made by BNFL, out of a total of 1062 fuel rods. During Cycle 9, iodine indicated some leaking assemblies. Fuel sipping after Cycle 9 found 8 or 9 leaking assemblies in Batch 9. They were taken out of service.

In November 1981, the licensee forwarded a final report on the cause of the 1979 fuel defects. The investigation was done by Battelle, Columbus Laboratories under an EPRI contract. They concluded that the following elements played a role in the failures: 1) fuel pellet chips caused high localized stresses in the cladding, and 2) the lower propensity of Batch 8 fuel to densify led to enhancement of fuel-clad contact pressure. A power change near the end of Cycle 7 may have played a role in causing the defects. Changes were made to the fabrication process to avoid pellet chipping, and refinements to the fuel design were planned for future batches, primarily an increase in the fuel-clad gap.

Haddam Neck submitted an amendment request to change the Technical Specifications to allow use of a revised fuel design developed to avoid fuel defects from the mechanism discovered in Cycle 8. A change to address concerns over operating with actual reactor core inlet temperature below its design value was included in the request.

License Amendment No. 52 was issued March 3, 1983, to revise the Technical Specifications to reduce the maximum allowable linear heat generation rate (LHGR) and adjust the axial power vs. offset curves accordingly. The changes were needed to allow use of a revised fuel design, developed to avoid the clad defect mechanisms observed in 1979. As discussed in Section 7.3 below, additional reduction of LHGR was imposed due to reactor operation with a lower than design core inlet temperature, which affected the calculated peak cladding temperature (PCT).

NRC's Safety Evaluation performed audit calculations to confirm the fuel design results submitted by Haddam Neck. The revised fuel design, run at reduced peak power levels, was found to be bounded by conditions previously analyzed and acceptable.

The changes were effective in minimizing fuel damage. Subsequently, fuel used during Cycle 9, 10 and 11 experienced progressively fewer clad defects, as indicated by reactor coolant iodine monitoring.

7.3 ECCS Performance

Haddam Neck had assumed that operation of the reactor with lower than design core inlet temperature was conservative with regard to the emergency core cooling system (ECCS) performance analysis. The licensee believed that peak clad temperature would decrease in a design basis accident if the core inlet temperature was decreased. This was not the case when the analysis was performed. The erroneous assumption had been accepted since Cycle 1, when core inlet temperature was reduced from its design value. The error was reported on December 11, 1981. However, the licensee did not provide the date of the temperature reduction.

As a corrective action to the reported error, the licensee proposed Technical Specification revisions to assure adequate ECCS performance as part of License Amendment No. 52. The ECCS analysis could meet PCT limits if certain conservative model assumptions were relaxed. However, the licensee found that the ECCS analysis would meet the Interim Acceptance Criteria limiting PCT to 2300°F, without relaxing the conservative model assumptions, by reducing LHGR. The staff accepted the analysis with the reduced LHGR.

7.4 EPRI Evaluation

In 1982, EPRI published an evaluation of stainless steel cladding for use in LWRs. It examined the performance of more than 550,000 stainless clad fuel rods used in six commercial power reactors located in the United States and Europe. Stainless steel clad for BWR fuel was inferior to zircaloy. In PWRs, however, stainless steel performance was comparable or superior to zircaloy.

EPRI found that stainless steel cladding had been widely used in the early years of nuclear power in a variety of facilities, such as power reactors, test reactors and ship reactors. However, zircaloy cladding provided better neutron economy and thus lower fuel costs. At the time of the report, only LaCrosse, Haddam Neck and San Onofre 1 continued to use stainless steel clad fuel in the U.S.

The report found that the performance of stainless and zircaloy fuel in normal conditions was similar. The response of the two materials to transients differed, depending on the transient considered. However, both materials were considered acceptable. Stainless steel had much higher permeability to tritium, which was reflected in higher tritium releases from plants that used stainless steel. EPRI concluded that the tritium release from the three U.S. LWR plants using it at the time of the report was not an environmental problem. Zircaloy had lower thermal neutron absorption, making it more economical since lower enrichment fuel could be used.

Of the six reactors examined for stainless steel fuel performance, three reported no defects. One reported two collapsed fuel rods but no other defects. Two, which include Haddam Neck's 1979 experience, reported incidents of approximately 50 to 100 leaking rods during a cycle but otherwise no defects. Fuel inspections were not always extensive. Nevertheless, reactor coolant iodine data support the assertion that very few rods stainless steel rods had cladding defects during operation of the plants, other than the incidents noted.

EPRI concluded that the performance record of stainless steel clad fuel was excellent. The performance was considered more significant because most of it had been achieved without any power maneuvering restrictions. The favorable results were attributed to the lower linear heat generation rate of PWR stainless steel fuel compared to zircaloy clad fuel.

7.5 Defects in Cycle 15, 1989

Cycle 15 began March 3, 1986 and ended on September 2, 1989. The operating cycle was followed by a 346-day refueling outage.

Reactor vessel internals work and fuel inspection and repairs accounted for the length of the outage. Fuel damage had been caused by machining debris left in the core after doing thermal shield work following Cycle 14. A number of metal chips got caught in the fuel, primarily at the bottom plate. Coolant flow caused the chips to rub against the fuel clad resulting in debris-induced fretting defects.

Cycle 15 experienced throughwall fuel cladding defects to 456 rods. Approximately 1500 additional rods sustained defects greater than 20% throughwall. Identification of the damage was complicated by the relative insensitivity of ultrasonic testing for detecting defects located at the bottom of the fuel rod. Additional testing methods were used to verify clad condition.

On December 15, 1989, the licensee reported 281 rods with throughwall defects in LER 50-213/89-20 on the basis of serious degradation of a principal safety barrier. The report may have been filed late, since documents show the licensee was aware of the damage as of October 19, 1989. No followup reports were filed as the extent of damage grew larger, eventually reaching a total of 456 failed rods. The NRC historical review team noted that the design basis for the waste gas decay tank (WGDT) rupture accident was 1% failed fuel, or 320 rods. No record has been located to demonstrate that the licensee recognized that this design basis had been exceeded. However, the design basis for the WGDT rupture accident also specifies the maximum amount of radioactive gases available in the event of a release. The licensee reported that the maximum curie content of the tanks over the period 1988 through 1989 was less than 5% of the design basis value.

The licensee had a fuel evaluation program in place during Cycle 15, but the evaluation method was not suitable for quantifying the number of failures due to the unique nature of the defects. The defects in the stainless steel clad occurred at the bottom of the rods, which limited the movement of water in and out of the rod due to gases trapped above the throughwall penetration. Water would enter the rod until the rod interior gas pressure equalized with reactor coolant pressure. The water remained in place unless a pressure or temperature change occurred. This resulted in significantly lower amounts of iodine in the reactor coolant than was usually observed when fuel cladding was breached. Noble gas concentration was considerably higher than usual, but this parameter was not used in PWR fuel evaluation procedures at the time, either at Haddam Neck or in the industry. The evaluation method used during Cycle 15 used iodine as the indicator and predicted 8 to 12 defective rods.

In response to the defects, the licensee conducted an extensive fuel inspection. Considerable effort was given to cleaning debris from the fuel and core, reconstituting fuel assemblies and improving the fuel monitoring program. In addition, the thermal shield was removed due to degradation of its support system.

The licensee devised a model to quantify throughwall defects during operation, which would indicate defects caused by the debris-induced fretting mechanism, by correlating Cycle 15 reactor coolant iodine and xenon measurements with the observed number of defects. Including xenon in the evaluation improved the accuracy of the estimated number of defects. The licensee presented calculations suggesting the method yielded defect estimates from 0% to 22% higher than the actual value.

7.6 Defects in Cycle 16, 1991

Cycle 16 began August 15, 1990, and ended October 17, 1991. The refueling outage lasted 149 days.

Because a certain amount of debris was expected to remain after the cleaning, and about 65 rods were expected to leak during Cycle 16, the licensee proposed amendments to its Technical Specifications that would limit the number of defects to 160 rods. This value was selected based on a steam generator tube rupture accident and was consistent with the Technical Specification limit of 1.0 μCi/g Dose Equivalent Iodine (DEI) activity. An action statement was included that required placing the reactor in hot shutdown if the estimated number of defected rods exceeded 160 rods for seven consecutive days. The proposed technical specification also included surveillance requirements to monitor the number of defected rods. A new basis statement was added which stated that a correlation method was the means to implement the surveillance requirement.

The proposed Technical Specification was issued on January 4, 1991, as license amendment No. 134. Prior to issuance, the licensee implemented the requirements through administrative procedures.

In March 1991, the plant went to Mode 5, cold shutdown, due to inoperable containment air recirculation fans. Reactor coolant iodine spiked to 1.78 μCi/ml, which exceeded reactor coolant specific activity limits. The only required action in Mode 5 was to increase the frequency of sampling until the DEI decreased below the limit. The event was reported in the annual report as required by the technical specifications. Prior to the shutdown, the fuel monitoring program estimated 25 defected pins. After the startup, the defected pin estimate spiked to 130 rods, then decreased to 30 defected pins.

On August 19, 1991, the plant began a shutdown as a precaution against the approach of Hurricane Bob. By the time power had been reduced to 40%, it was clear the hurricane would bypass the site, and the plant returned to full power. The xenon spike that occurred after the maneuver caused the indicated number of throughwall fuel rod defects to increase to 418, although I-131 did not exceed 0.01 μCi/ml. The licensee projected that it would exceed the seven-day LCO, and met with NRC staff on August 26, 1991, to discuss the issue. Haddam Neck personnel presented evidence that the spike was similar to others observed during power maneuvers, and that an alternate estimation method, based on those examples, could be used to better evaluate the spike on August 19. CYAPCo asserted that the LCO did not apply because alternate estimation methods yielded lower numbers. The NRC staff did not object to that assertion.

The Haddam Neck control room log recorded exiting the LCO on August 28, 1991, within the allowed seven-day period, on the basis of a plant chemistry report that the defective fuel estimate decreased to 95 rods using an alternate estimation method. However, the defective fuel estimate based on the method approved in the Safety Evaluation for the applicable Technical Specification did not decrease below 160 rods until September 1, 1991, about 10.5 days after the first indication that the LCO had been entered.

CYAPCo performed a safety evaluation of the change to their defected fuel estimation procedure prior to applying it to the surveillance. The revised procedure used a graphical method to plot the number of fuel clad defects projected to exist in ten days. It allowed the alternate method to be continued for ten days before concluding that the number of defects had changed. The plant staff concluded that no unreviewed safety question was involved and no change was required to the Technical Specifications. That may have been erroneous. The revised method appears to have been less conservative than the method specified by Technical Specifications since it yielded a lower value. Thus, the change may have reduced the margin of safety, which fits the definition of an unreviewed safety question. In addition, the basis of the Technical Specification described the surveillance method to be used to comply with the requirement. Changing the method may have been a change to the Technical Specification basis.

A fuel inspection done after Cycle 16 estimated that 102 rods were defective. It is not clear from available records if the inspection examined all the fuel assemblies. However, the revised fuel performance program indicated about 100 defected rods at shutdown, which agreed with the number of defects found.

Starting with Cycle 17, the licensee began changing to zircaloy clad fuel. The conversion was complete, except for 5 assemblies, by Cycle 19.

The licensee's Semiannual Radioactive Effluent Release Reports that reported the effluent impact from these fuel defects included a summary of the quantities of radioactive liquid and gaseous waste effluents, including any unplanned or abnormal releases, a summary of meteorological data associated with the gaseous effluents, an assessment of the radiation doses from the radioactive liquid and gaseous effluents released to the environment, quantities of radioactive waste disposed of and changes to the Radiological Effluent Monitoring and Offsite Dose Calculation Manual. The Semiannual Radioactive Effluent Release Reports contained plant data in accordance with the guidance in NRC Regulatory Guide 1.21, Revision 1, June 1974. The reported radioactive effluent releases were within the quantities projected in the Final Environmental Statement, which was issued by the Atomic Energy Commission in October 1973. The reported effluents and the associated calculated annual doses to a member of the public were in accordance with the ALARA criteria of Appendix I to 10 CFR Part 50. The licensee used accepted NRC methodology contained in Regulatory Guide 1.109 for the calculation of annual doses to man from the release of radioactive material in nuclear power reactor effluents.

These reports noted that the solid waste streams had transuranic isotopes, which resulted in primary side spent resins often being classified as Class C. (Most power reactor waste streams are typically Class B waste.) However, the effluent and environmental releases were not impacted from the fuel defects, except for one quarterly Technical Specification limit following the 1989 fuel failure. Annual exposure limits were not exceeded.

Conclusions

Haddam Neck stainless steel fuel received four major evaluations from AEC and NRC between 1967 and 1983. The design was acceptable on each occasion and was bounded by conditions at the San Onofre Unit 1 reactor.

The Cycle 8 (1979) fuel defects were due to manufacturing defects, exacerbated by a power ramp performed near the end of Cycle 7. Power ascension limits combined with fuel design changes were effective in minimizing fuel cladding defect formation due to manufacturing defects.

More than 1% of the fuel rods had defects at the end of the Cycle 15 (1989). This value exceeded the design basis value for fuel rod defects found in the FSAR analysis of the waste gas decay tank rupture accident. However, the actual curie content of the tanks was less than 5% of the design basis value assumed for calculating off-site dose consequences.

The licensee's safety evaluation of the change to the failed fuel estimation procedure used in Cycle 16 (1991) may have been in error when it concluded that no unreviewed safety question existed and no change was needed to the technical specifications. If either condition existed, prior NRC approval would have been required to make the change.

The results from operating with defected fuel were a gaseous release exceeding the quarterly RETS limit in 1989. Also in 1989, positive levels of I-131 were detected in vegetation samples taken near the site boundary. The 1989 release did not exceed 10 CFR Part 20 exposure limits. Overall, the liquid and gaseous radioactive waste effluent data were properly documented and reported in accordance with 10 CFR 50.36a and Criterion 60 of Appendix A to 10 CFR Part 50. No significant errors or omissions were identified. The calculated annual doses were in accordance with the ALARA criteria of

Appendix I to 10 CFR Part 50. Another result of the fuel defects was alpha contamination of the interior surfaces of plant equipment.

Concerns to be addressed during site characterization include characterization of alpha contamination of plant primary and secondary systems to determine appropriate procedures for dismantlement and worker protection. Although environmental data do not indicate significant transurancies (indicated by Am-241 gamma), characterization of the site should take the potential for alpha contamination into account.

SUPPLEMENT A-1

Historical Site Assessment Data Table

Date:	CY Doc #:	Location:	Survey Area #:	Event Type:	Medium:	Event Description:	Corrective action:
5/3/69	AO 69-6	Tank 1A / surrounding yard asphalt		Red	200 gal. distillate	200 gal. distillate spill from crack at disked head seam	Flush area w/ 20,000 gal. of water. Run off to storm drn.
5/6/69	AO 69-7	Floor area boron evaporator		Rad	Waste liquid	500 gal. spill from boron evprtr. via flr drains to aerated drain tanks	Decon.
1/1/72		Discharge Canal		Red	Soil	Dredge Canal	Dredge Canal to estabilsed criteria
1/15/73	PIR 73-18	RWST		Red	Liquid	RWST Thermo siphon heater leak. Primary to secondary	Isolated heater. Install new headgasket.
1/17/73	DEP Notif.	CT. River		Red	Liquid	2.86x10/4 gal/day	Unknown
3/28/73	PIR 73-111	Diesel Gen. flr drains, #2 tunnel		Non Rad	Diesel Fuel Oil	#2B Diesel Gen. Fuel Oil Day Tank overflows during filling	Clean up oil in #2 canal. Investigate alarm usage.
4/6/73	LTR	Floor drain to Discharge tube		Non Rad	Diesel Oil	Overflow of tank	Unknown
4/26/73	PIR 73-134	Steam Generator		Red	Gas	High RMS Alarm - Air Ejector Monitor R-15	Identify leaking steam generator. Plugged tube in SG #1
5/18/73	PIR 73-154	"A" Aerated Drains Tank		Red	Liquid	Abnormal inleakage to ADT from open SFP filter drain valve.	Valve closed. Construction personnel cautioned.
6/21/73	AO 73-6	Ion exchange cubicles		Red	Liquid	500 gal. leaked to the aerated drains tank.	Replace diaphragm valve
6/21/73	AO 73-7	PAB pipe trench		Red	Distillate	20 gal. spill from valve to pipe trench of PAB	Unknown
11/1/73	AO 73-11	RWST		Red	Liquid waste eff.	270 liters of liquid released to storm drain,diluted w/ 6000 gal.of service water.	Replace valve diaphragm
4/18/74	PIR 74-70	Discharge canal, tunnel, Flr drains		Non Rad	Mobil DTE 797 Oil	100-500 gal. oil spill from Turbine bearing seal oil cover	Spread oil pickup material, plug flr. drains
4/26/74	AO 74-7	PAB		Red	Radioactive gases	Leaking seal on volume control tank hydrogen regulator causes gas rel.	Regulator replaced
6/9/74	PIR 74-108	"A" Test Tank		Red	Liquid	2 gal. spill from hole on top of "A" TT during transfer from "A" ADT.	Stopped Transfer. Vent holes plugged.
10/15/74	PIR 74-180	Steam Generator Blowdown Tank		Red	Liquid	SG Blowdown Tank Rupture Disk actuates.	Replace Rupture Disk
12/30/75	PIR 75-204	Steam Generator Blowdown Tank		Red	Liquid	SG Blowdown Tank Rupture Disk actuates.	Replace Rupture Disk
1/28/76	PIR 76-15	"A" Recycle Test Tank		Rad	Liquid	15 gal. of water leaks from "A" Test Tank to diked area.	Isolated and tagged line. Submitted MR.
2/27/76	PIR 76-30	RWST Thermosyphon Heater Valve		Red	Liquid	5 gal. leak from TSH valve.	Secured leak. Submitted MR.
6/15/76	PIR 76-66	Below drumming room floor		Red	Contaminated soil	Waste discharge pipe erosion allows spill below drumming Rm flr.	Weld saddle over defective area. PDCR pending
6/29/76	PIR 76-79	"B" Waste Gas Decay Tank		Red	Gas	Rupture disk actuates after switching from "A" WGDT to "B" WGDT	Isolated "B" WGDT.
12/14/76	PIR 76-138	Below drumming room floor		Red	Contaminated soil	Waste discharge pipe erosion allows spill below drumming room flr.	Leak repaired.Contaminated sand drummed.Imp. PDCR
12/20/76	PIR 76-140	RWST Thermo Syphon Heater		Red	Liquid	Spill from weld to blacktop. An area 4' in diameter was affected.	Pipe weld repaired.
12/29/76	PIR 76-142	"A" BWST well and diked area.		Red	Liquid	"A" BWST heater leaks to heater well and diked area.	Isolated Heater. Submitted MR
2/24/77	LER 77-2	"A" Recycle Test Tank manway		Red	Liquid	1000 gal. of radioactive water released to diked area around tank	Temporary clamps applied, leakage stopped.
6/9/77	PIR 77-45	115KV Switch Yard		Rad	Particulate	Particle discovered during routine survey	Particle removed and analysed. Area Surveyed without incident
7/29/77	PIR 77-82	Weir Box		Red	Liquid	Background increase detected above Weir Box	De-sludge Weir. Fabricate shield over Weir.
9/6/77	PIR 77-83	Spent Fuel Building Sump		Red	Liquid	5400 gal. to RTT to RPWT, to SFB sump to ADT, to BWST 2400 gal. to Rvr	Procedure change.
1/30/78	PIR 78-23	"A" Recycle Test Tank		Red	Liquid	Unknown amount of liquid spills from tank to sump	Unknown
1/31/78	PIR 78-24	RWST		Red	Liquid	Spill fromThermosyphon Heater	Isolated heater and replaced head gasket
2/10/78	PIR 78-25	East of the W.N. tank		Non Rad	Sodium Hydroxide	5 gal of caustic passed through broken flange at W.N. on to floor	Unknown
3/17/78	PIR 78-33	Below drumming room floor		Red	Contaminated soil	Waste discharge pipe erosion allows spill below drumming room floor	Weld leak area
11/25/78	PIR 78-120	BWST		Red	Liquid	Leak from the thermosyphon heater contaminates secondary system	Isolate steam to heaters. Drain condensate receiver
11/29/78	PIR 78-122	Outside Yard		Red	Particulate	Guard reports shoes contaminated after rounds on Hot side (BWST)	Roped off area, notified HP, survey of Guard House
1/10/79	PIR 79-04	RWST Heater Isolation Valve		Red	Liquid	Valve diaphragm ruptures causing steam and water leak from hole in handwhl	Isolated steam. Plugged hole
2/17/79	PIR 79-27	Stack to yard via manway		Red	Liquid	Rupture disk activates, 20 gal. stack to yard via manway.	Area roped off, frozen matl. to 55 gal.drm, wash to drain
3/6/79	PIR 79-38	Main Stack Hatchway		Red	Liquid	Rupture diaphragm activates. Water leaks from hatch .	Isolate Steam Generator blowdown valves.
5/9/79	PIR 79-63	BWST		Red	Liquid	Sampling indicated that the "B" BWST siphon heater was leaking	Siphon heaters isolated. Condensate receiver dumped
5/17/79	RPT 79-14	CT. River		Non-Rad	Chlorine	Chlorine discharge of 360 min. vs 120 min. allowable	Mod. of piping system to ensure relief flow back to tank
6/4/79	PIR 57-79	Fuel Pool ventilation duct		Red	Liquid	Fuel pool water overflowed into ventilation duct during cavity drain operation	Procedure clarification. Personnel counselling
7/23/79	PIR 79-83	Main Stack Area		Red	Liquid	Liquid dripping from ventilation system flange to main stack	Decontaminate Area
8/10/79	PIR 79-92	Hot machine shop driveway		Red	Contaminated Soil	Disjointed section of piping causes spill during excavation, cont. soil	Decontaminate Area
9/29/79	PIR 79-105	Ventilation duct, PAB and Yard		Red	Liquid	50 gal. spill to RCA yard, 500 gal. to PAB floor	See CR 79-806
11/27/79	PIR 79-126	CY/Millstone Point		Red	Particulate	Contaminated staging sent to MP from CY. Lack of material control.	Unknown
11/28/79	PIR 79-122	"B" BWST Syphon Heater		Red	Liquid	Contaminated water enters feed and condensate system.	Two leaking tubes were plugged. WP MA 0816
12/16/79	LTR 3/16/80	Contamination, Roof of PAB		Red	Liquid	Contamination on PAB roof is fixed by asphalt base sealer	Liquids cleaned up,catchbasins pumped out,area decon.
12/16/79	PIR 79-125	Stack, Yard Drains		Red	Liquid	Degasifier fills w/ reactor coolant. Spill at stack and yard drain hold tank	See PORC Meeting Minutes #79-97
1/20/80	LER 80-15	Waste disposal bldg floor		Red	Liquid	6" of water found on waste disposal bldg. floor	Unknown
2/3/80	LER 80-05	"B" BWST		Red	Liquid	Leak in "B" BWST Thermo Syphon heater	Replace gasket
2/7/80	LER 80-05	"B" BWST		Red	Liquid	Leak in "B" BWST Thermo Syphon heater	New lower head fabricated and installed
2/14/80	PIR 80-26	Yard storm drain (Drumming Rm)		Red	Liquid	Activity found after sampling the yard storm drain	The drain was pumped down to the ADT
2/27/80	PIR 80-33	"A" BWST		Red	Liquid	Leaking flange form piping fills BWST diked area with 3 inches of water.	Flange receives new gasket. Heater returned to service.
2/27/80	PIR 80-34	Below drumming room floor		Red	Contaminated soil	Contaminated sand was found in the river effluent pit	Drain Line sealed. Sand remediated.
3/10/80	PIR 80-37	Facility Grounds		Red	Particulate	Contamination found in uncontrolled areas. Roofs and grounds.	Areas decontaminated. Contamination Fixed
4/19/80	PIR 80-54	Main Stack Duct		Red	Gas	12" X 2" split in main stack duct seam leaking to atmosphere	Replace Duct.
4/29/80	PIR 80-58	Yard drain No. 4		Red	Liquid	Activity leached out from mud in bottom of yard drain	Drain pumped out, flushed and mud removed
5/3/80	EN-149	Stack		Red	Gas	Venting VCT to Waste Gas Surge Tank received RMS. alarm	Set point change to waste gas purge tank valve
5/4/80	EN-149	Stack		Red	Gas/liquid	Water/steam release from blowdown tank through cont. ductwk & stack	Design change/Procedure revision
5/4/80	PIR 80-62	Main Stack		Red	Liquid	Steam Generator blowdown tank rupture disk actuates. steam exits stack.	Procedural revision.
5/19/80	PIR 80-72	CT/Hope Creek Station		Red	Components	Lack of material control. Turbine inner casing shipped w/o HP release.	Require HP approval prior to Red matl shipments.

Historical Site Assessment Data Table

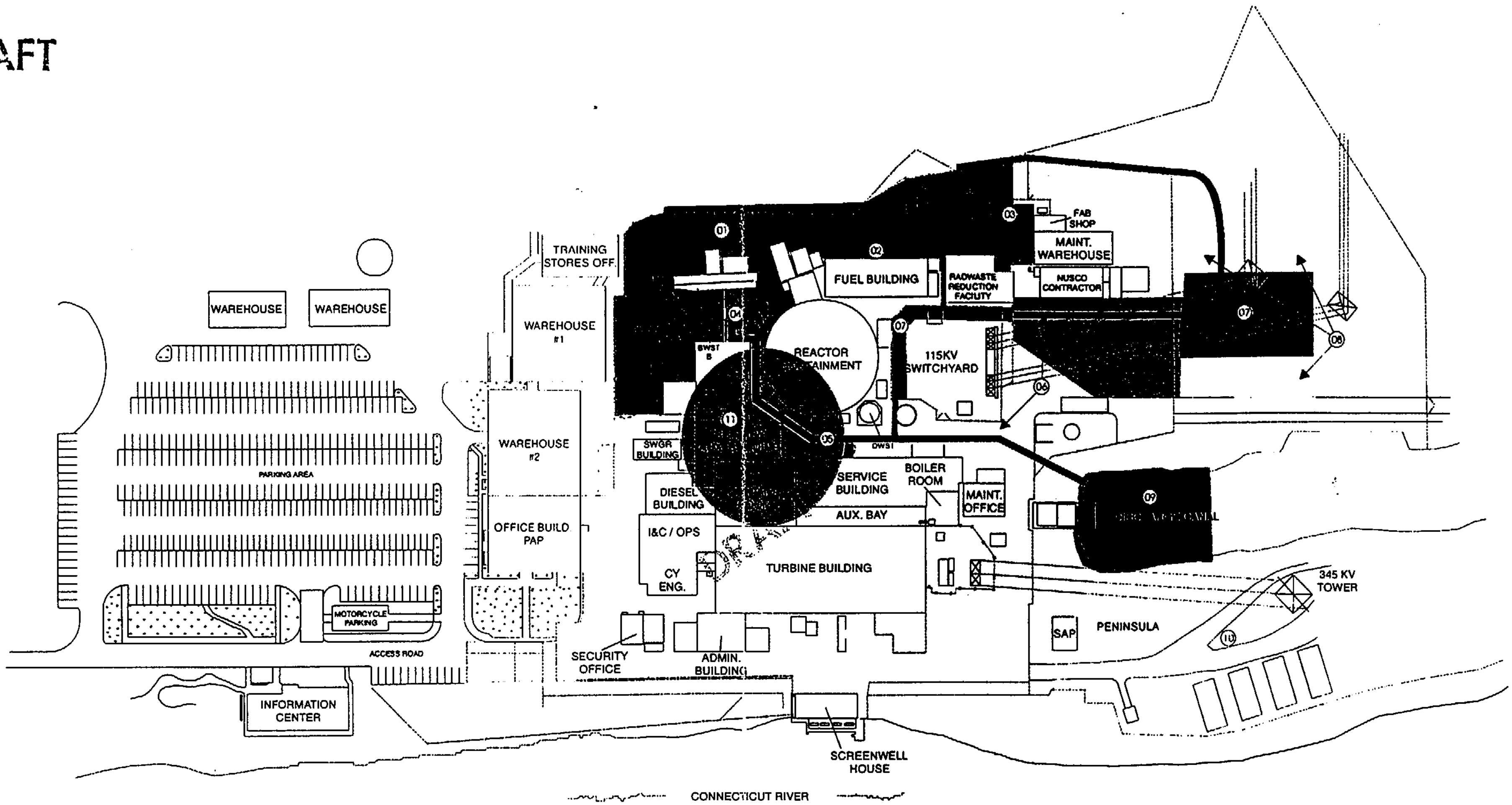
Date:	CY Doc #:	Location:	Survey Area #:	Event Type:	Medium:	Event Description:	Corrective action:
12/8/80	PIR 80-143	"B" BWST Syphyon Heater		Rad	Liquid	Spill from "B" Boron Waste Storage Tank Syphon heater to diked area.	Replace gasket. Drain Dike area to ADT.
2/9/81	PIR 81-10	Administrative Building		Rad	Particulates	Contamination found at three locations.	Decontaminate areas.
4/22/81	PIR 81-38	CY/ North Haven		Rad	Components	Lack of material control. Contaminated reheater tubes shipped off site.	Procedural changes. Material inventory.
7/25/81	PIR 81-82	Steam Generator		Rad	Particulate	Primary to secondary leak in steam generator # 2.	Unknown
9/17/81	LER 81-15	Atop primary auxiliary building		Rad	Gas	Leak from exhaust duct to main stack	Repairs complete . Permanent repairs during outage.
9/17/81	PIR 81-98	Ductwork on PA Roof		Rad	Gas	A 3' section of ductwork seam on the PAB exhaust fan failed.	Repair during refueling.
9/22/81	PIR 81-101	RWST		Rad	Liquid	15/20 gal leaks from the RWST hatch	Repairs complete
10/3/81	PIR 81-106	RWST		Rad	Liquid	Activation of heater causes RWST to overflow to diked area.	Pump RWST to SFP. Isolate Steam to heater.
10/4/81	HP Survey	RCA at rails SFB LL		Rad	Liquid	RCA sample at rails after spill from SFB lower level.	Area Decontaminated
10/19/81	PIR 81-114	PAB Ventilation System		Rad	Liquid	Inoperative Waste Liquid Evaporator level indicator causes contamination.	Decon. PAB vent. area and return level indicator to normal.
12/1/81	RPT	CT. River		Non Rad	Sodium Hypochl.	3800 gal. spill from storage tank drain valve	Unknown
12/31/82	PIR 82-108	BWST valve manifold		Rad	Liquid	Contaminated liquid found by BWST manifold.	BWST dike drains opened and the thermosyhon heater isolated.
3/28/83	PIR 83-37	Septic Tank		Rad	Liquid	84 gal. Contaminated H20 from Chem. Lab to Septic Tank	Drain opening bolted shut. Door to drain room secured.
4/14/83	PIR 83-47	Steam Generator #2		Rad	Liquid	Primary to secondary steam generator tube leak.	Repair leak.
4/18/83	PIR 83-42	Hot side flr. drain		Rad	Liquid	Yard drain to manway # 5	Drain and manway pumped down
7/28/83	PIR 83-75	RCA yard manway & #5 yard drain		Rad	Liquid	RCA manway overflows to RCA yard and enters # 5 yard drain	Pumped out RCA manway and yard drain #5
7/31/83	PIR 83-78	BAMT overflow pipe, 2-3 Ft. Radius		Rad	Particulate	Rust & scale from BAMT overflow pipe contaminate personnel shoes	Post area and decontaminate
12/13/83	PIR 83-137	"A+B" RTT		Rad	Liquid waste eff.	400 gal. release via mis-positioned valves	Cautioned operators. Procedure revised
4/3/84	PIR 84-39	Ground between "B" BWST dike & RVST		Rad	Liquid	20 gal. liquid spill between "B" BWST and RVST	Pump water from "B" BWST to "A" BWST. Clean-up H2O
5/25/84	PIR 84-72	RCA Yard and Chem. Lab.		Rad	Liquid	Chem. Lab drain overflows,ADT drain line plugged causing flow to yard	Yard & Lab. cleaned. Yard drain #5 coated and routered
9/11/84	PIR 84-181	Resin pit, Resin sump pump		Rad	Resin	Resin liner overflowed	Resin removed from pit and sump pump. Area deconed
9/13/84	PIR 84-182	Resin liner and area		Rad	Resin	Resin liner overflowed	Unknown
11/6/84	PIR 84-244	Condensate Receiver,Secondary side		Rad	Liquid	Reduction in steam pressure allows water to enter condensate receiver	Clean-up by feed and bleed
11/13/84	PIR 84-242	A and B Boilers		Rad	Liquid	Condensate receiver contamination verified by Chemistry.	Investigate pathways for contamination and repair.
11/26/84	PIR 84-264	RCA yard, yard drains		Rad	Liquid	Water from RCA Yard drain cover with possible path to storm drain	Operators cautioned about proper valve positioning
3/20/85	PIR 85-51	Yard drains #4 and #5.		Rad	Liquid	Drains 4&5 show activity via Adams fltr. dike via cond. relief valve via #2 SG	Pump down YD, Drain plug in Adams filter dike area replaced
3/25/85	PIR 85-52	Yard Drains		Rad	Liquid	ADT drain culvert overflows to RCA yard and yard drain #5	Pump down YD 5. Clean of drains performed.
5/17/85	PIR 85-74	Yard drains No. 4 & 5		Rad	Liquid	Yard drains # 4&5 show activity from leaking Stm. Gen. BD valve	Pump down yard drains #4 and #5.
6/3/85	PIR 85-84	Radwaste reduction facility		Mixed	Sodium Hydroxide	3 gal. spill of sodium hydroxide during radwaste presolidification operation	Spill and area cleaned-up.
1/22/86	PIR 86-23	Ground adjacent to WG Bldg. wall		Rad	Liquid	Broken drain connection on temp. drain line drips to ground	Absorbent material laid down on spill. Pipe repaired
11/13/86	RA-1142	Discharge Canal		Rad	Soil	Dredge Canal	Dredge Canal to estabilised criteria
6/26/87	PIR 87-67	Black Top		Non-Rad	Demin. Water	140,000 Gal. flowed across black top to discharge canal. Truck hits PWST	Install protective barriers
8/2/87	87-CY-7151	Containment access area		Rad	Liquid	Drain hose spill	Decon.
8/7/87	87-CY-7151	Reactor Equip. Hatch Moat		Rad	Liquid	Water spill at equipment hatch moat	Unknown
4/7/88	PIR 88-81	Steam Generator #2		Rad	Liquid	Primary to secondary leak. S/G #2 tube leaks compromises secondary side	Chemistry to monitor leak rate.
4/12/88	PIR 88-89	Service Bldg hallway floor		Non- Rad	Fuel oil	25-50 gal. Fuel oil spill to sand when line cut by saw	35 sq. ft.of oil soaked soil was remediated
9/30/88	PIR 88-176	Access Road		Non-Rad	Hydraulic Oil	25 gal. spill from backhoe onto access road.	Contain and cleanup spill
10/11/88	PIR 88-181	Yard drain number 1		Rad	Contaminated Soil	1200 Ft/3 of contaminated soil was discovered around MH No. 1	Periodic cleaning of drain lines to the ADS
12/16/88	PIR 88-218	Chem. Lab drain sump		Rad	Liquid	Leak from corroded cast iron pipe to sump	Replace iron pipe with plastic
2/24/89	PIR 89-35	Leach field 115kv yard		Rad	Liquid	50 gal. spill of rad. liquid from SFB Flr. drain, line discharges to 115KV yard	See memo W. Pomberg to E. Mroczka
7/25/89	*PIR 89-110*	*Hypochlorite Room*		Non-Rad	Hypochlorite	50 gal. spill of hypochlorite from broken line	See memo E.Mroczka to W.Pomberg
10/12/89	PIR 89-174	RHR pit		Mixed	CC water	Unknown volume of water spilled from flange in RHR pit	Unknown
3/13/90	PIR 90-44	Screenwell house		Non-Rad	Hypochlorite	600 gal. of hypochlorite release to river from cracked pipe	Secured source
3/16/90	PIR 90-48	Spill from loop #3 fill valve bonnet.		Rad	Liquid	After returning loops 2, 3, and 4 to the drain header water spilled from FVB.	Isolated spill. Deconed area.
3/22/90	PIR 90-52	RCA yard area storm sewer		Mixed	CC water	330 gal. spill of component cooling water from cracked piping	Isolated the CCW supply
4/10/90	PIR 90-64	Discharge canal		Non-Rad	Fuel oil	Unknown volume of oil released into discharge canal	Solid tops supplied for NPDES Sump(+ North & South)
4/11/90	PIR 90-65	RCA Yard		Rad	Liquid	Contaminated hoses dropped. Service Bldg. to containment blocked	Removed source, isolated area, decontaminate
8/16/90	PIR 90-203	Steam Generator #2		Rad	Liquid	Primary to Secondary steam generator leak.	Repair during outage.
9/3/90	PIR 90-213	Steam Generators # 1, 2, and 3.		Rad	Liquid	Primary to secondary leak.	Unknown
9/14/90	PIR 90-239	RWST		Rad	Liquid	RWST shows signs of a 6 gal. per day leak.	Repair during outage.
8/12/91	PIR 91-149	Pipe Trench		Rad	Liquid	400 gal. spill from RCS via open valve to pipe trench.	Safety valve handles to be installed.
11/17/91	PIR 91-278	RWST		Rad	Liquid	RWST flange leaked when RWST was being filled during cavity pump down.	Stopped pumping cavity. Personnel counselled on quality wk.
1/21/92	PIR 92-27	RWST		Rad	Liquid	Leak RWST to diked area. Chem. sampling H2O different from current H2O	Contain water in dikes. Monitor and evaluate leak rate.
4/15/94	PIR 94-076	PAB Floor Drains		Rad	Liquid	Discharge from PAB supply heat exchanger enters Adams drain to yard drain	Seal drain. Identify drains according to use.
7/21/94	PIR 94-125	RWST		Rad	Liquid	RWST leak produces standing water	Seal Yard Drain. Initiate Monitoring.
11/23/94	ACR 94-179	Vendor		RAD	Components	Gages with fixed contamination were inadvertently released from the plant site.	Material retrieval. Awarness training. Procedural change
2/12/95	PIR 95-067	Yard Drains #4, #5, #6		Rad	Liquid	During sampling activity was found in the yard drains.	Barrier Plugs to be installed.
8/1/95	ACR 95-250	115kv switchyard		Rad	Fixed cont.	Fixed contamination found outside RCA in the 115KV switchyard.	Remove cont. pavement & dirt. Frisk to < 100 cpm
11/15/95	ACR 95-472	RCA Yard		Rad	Liquid	Lack of material control. Contaminated hose used for hydrolazing.	Check all hoses prior to use on clean systems or equipment
2/13/96	ACR 96-0158	Warehouse		Rad	Components	Lack of material control. Contaminated filters found in warehouse from '79	Survey filters and store in RCA.

Historical Site Assessment Data Table

Date:	CY Doc #:	Location:	Survey Area #:	Event Type:	Medium:	Event Description:	Corrective action:
5/20/96	ACR 96-559	Dumpster behind machine shop		Rad	Equipment	Bomb detector equipment containing source found in trash dumpster.	Provide quidence on proper disposal
10/7/96	ACR 96-1185	Yard drains #1, #4, #5 and #6.		Rad	Liquid	Tritium found in yard drains. Probable cause is leak in RWST to Yard drain #1	Repair RWST as soon as possible.
11/30/96	ACR 95-509	"Waste Gate"		Rad	Particulate	During a routine survey a "particle" was located in the asphalt.	Continue boundary surveying.
2/27/97	ACR 97-104	Vendor		Rad	Equipment	Contaminated material inadvertently released to an unlicensed vendor.	HP to survey all materialleaving RCA. Conditional Release Eliminat
2/28/97	ACR 97-106	Steam Generator Mock-up Building		Rad	Equipment	Contaminated equipment found outside The RCA	HP to survey all materialleaving RCA. Conditional Release Eliminat
7/17/97	ACR 97-450	Rifle range and adjacent landfill		Rad	Particulate	Contaminated material found outside the protected area during Site Survey	Area secured. Security patrol ea. shift. Det. extent of Cont.
8/21/97	ACR 97-0670	RWST		Rad	Soil	Activity located in sand by RWST. Uncontrolled release path.	Repair barrier, limit activity. Repair RWST.Remediate
8/29/97	ACR 97-0694	Closed Loop Cooling Water System		Rad	System	Activity detected in "clean" Closed Loop Cooling Water System	Est. Sampling Sch. Mark drum. Sec. potential release paths.
9/3/97	ACR 97-0713	Vendor		Rad	Liquid	Potentially radioactive water was shipped to a Hazardous Waste vender	
9/4/97	ACR 97-0716	Lowland area by SE gate		Rad	Liquid	Tritium located outside fence at discharge point. Unmonitored release path	Eliminate release path. Perform saftey eval. Decon.
1973-1994		Various		Non-Rad	Gas,Oil, Hydraulic	Various small spills	Invstg reportabillity,disposal of drums,dose to public,vendor lic.
1981-1983		Steam Generator		Rad		Various tube leaks	Repair during outage

SUPPLEMENT A-2

DRAFT



- General area of site affected by radiological events
- Yard drains affected by documented radiological events
- Area of potential migration of residual radioactivity
- Potential residual radioactivity in the expanded protected area. (orig grade)
- Area of potential residual radioactivity from sub PAB floor effluent piping leaks

EVENTS ASSOCIATED WITH AREAS OF SUSPECTED RESIDUAL RADIOACTIVITY

1) Resin transfer and related activities resulting in yard area surface contamination
2) Yard area surface contamination outside of Fuel building
3) Radwaste storage and handling/RCA run-off
4) Tank leaks, Tank heater leaks and Primary vent stack events
5) Contaminated drain system overflow release caused by system backup
6) Radioactivity identified in Yard Storm Drain system effluents
7) Fuel building drain event
8) Migration flow path of radioactivity from the RCA or affected areas
9) Accumulation of radioactivity from storm drain and monitored release effluents
10) Area affected by storage and handling of material and equipment
11) Leakage from piping of monitored effluent release path under PAB

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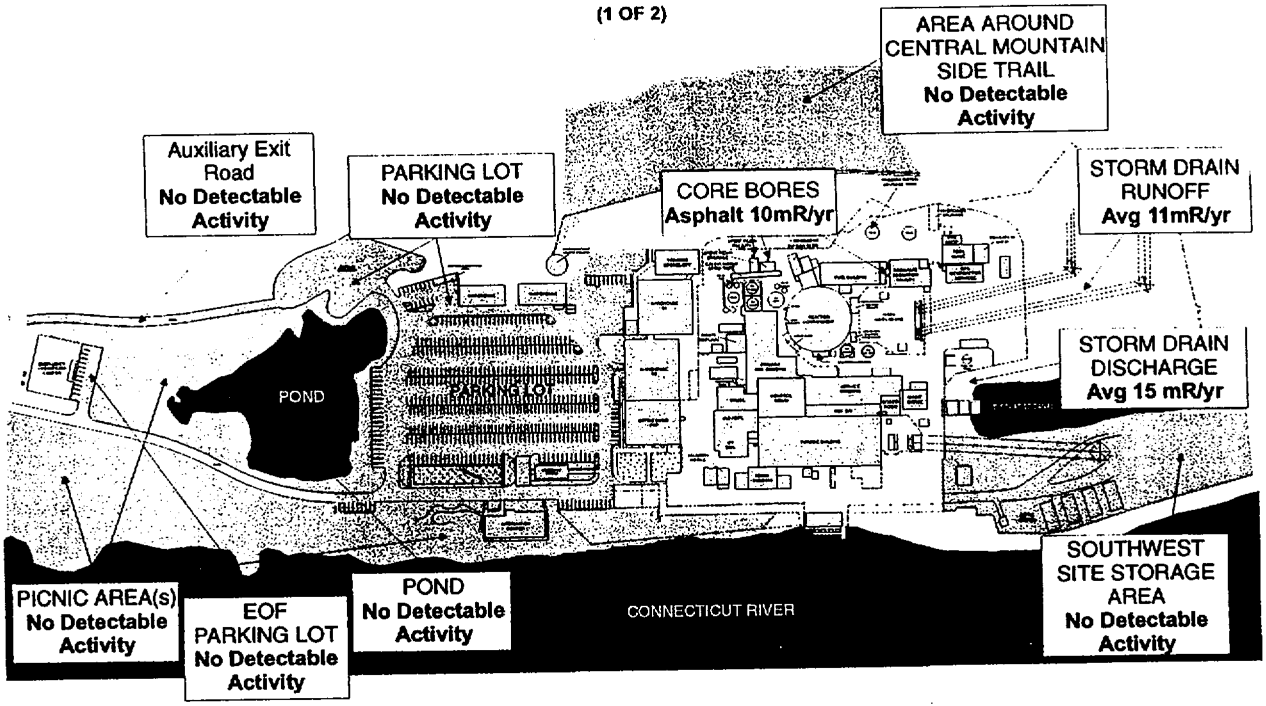
SUPPLEMENT A-3

SCOPING SURVEY STATUS
(1 OF 2)



INDICATES AREAS OF THE SITE THAT HAVE BEEN SURVEYED / SAMPLED FOR RADIOACTIVE MATERIALS - NO DETECTABLE ACTIVITY FOUND

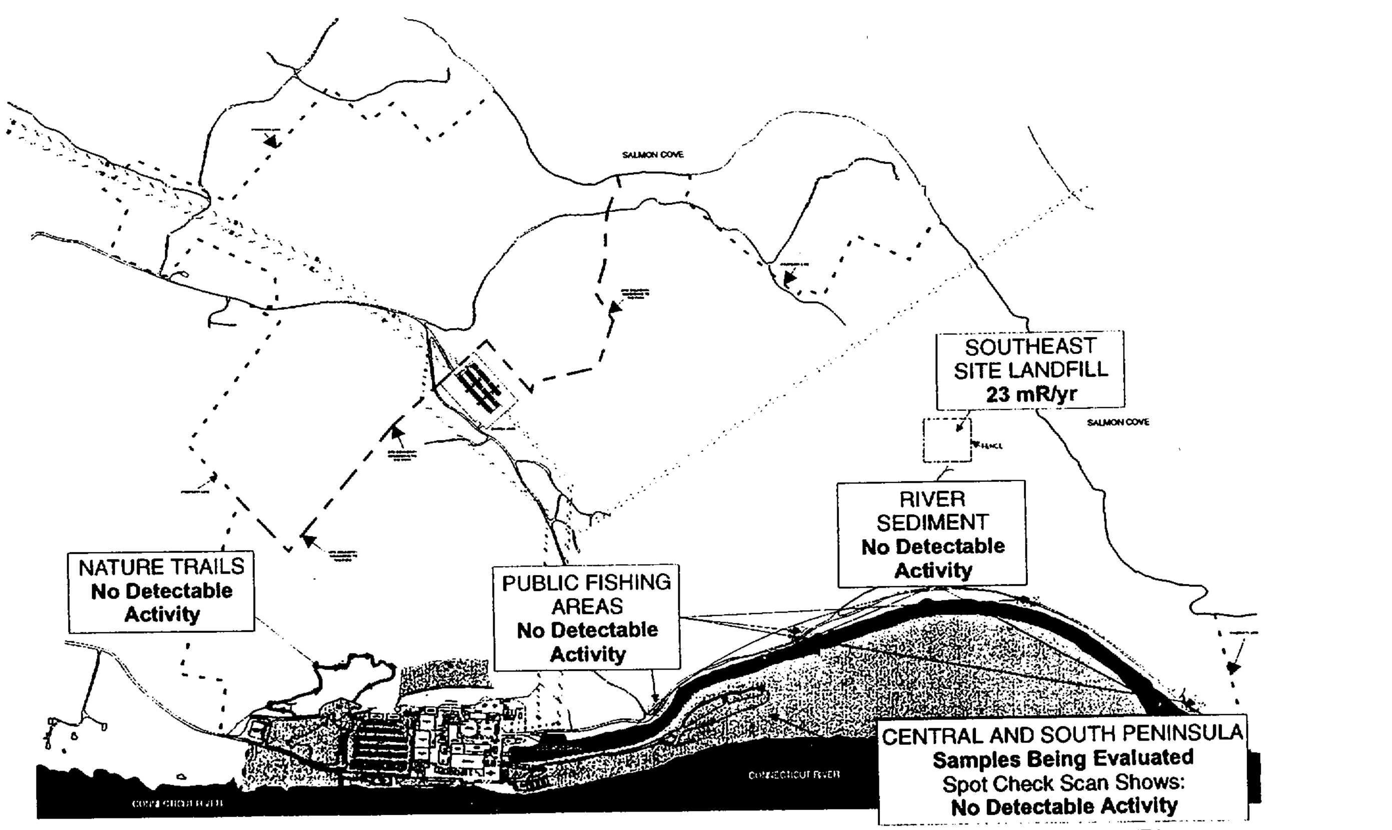
INDICATES AREAS OF THE SITE THAT HAVE BEEN SURVEYED / SAMPLED FOR RADIOACTIVE MATERIALS - MEASURABLE ACTIVITY FOUND

COMMENTS: Areas Evaluated by Scanning, Soil Sampling and Exposure Rate Readings

Revision #3 10/20/97

Revision #3 10/20/97

SCOPING SURVEY STATUS
(2 OF 2)



INDICATES AREAS OF THE SITE THAT HAVE BEEN SURVEYED / SAMPLED FOR RADIOACTIVE MATERIALS - NO DETECTABLE ACTIVITY

INDICATES AREAS OF THE SITE THAT HAVE BEEN SURVEYED / SAMPLED FOR RADIOACTIVE MATERIALS - MEASURABLE ACTIVITY FOUND

APPENDIX B

NRC RESPONSE TO SIGNIFICANT RADIOLOGICAL OCCURRENCES AND OPERATIONAL EVENTS

INTRODUCTION

The NRC response to significant radiological occurrences and operational events at the Haddam Neck site is discussed in this portion of the report. The team first reviewed the licensee's reports to determine what was known by the licensee and what was reported to the NRC. After the scope and extent of the history were determined, the team reviewed NRC inspection reports and licensee correspondence to determine the NRC response to the occurrences and events. A short section is included to define the general standards and guidelines used by the NRC inspectors for response to these types of occurrences and events.

1. LICENSEE REPORTING OF RADIOLOGICAL EVENTS

Scope

In an effort to understand the scope and extent of the history at Haddam Neck related to spills and contamination, the team reviewed the licensee's internal and external reports and notifications regarding events in the areas of radiological spills/releases, problems with stainless steel fuel cladding, radioactive waste system occurrences and release of potentially contaminated materials from the facility. The documents covered the period from 1967 to present, with an emphasis on reports required by 10 CFR 50.73.

Details

1.1 Internal Plant Reports

The licensee used the Plant Information Report (PIR) system, and later, the Adverse Condition Report (ACR) system, as an internal reporting system to identify and track any condition in the plant that required follow up or corrective action. In addition to plant operational events, the systems were also used to document spills and unplanned releases to the environment.

PIRs and ACRs related to potential contamination events were assembled by the licensee for its own review to aid in the site characterization for decommissioning planning/preparation. The recently assembled documents covered the period from 1967 to present and were made available to the NRC team. Events, releases and spills were recorded for the entire operating period of the plant, from start-up through the time of shutdown. Major events had significantly more detail written in each report. The reports generally reflected the information included in NRC Region I Inspection Reports. There were no events in NRC Inspection Reports that were not captured in the licensee's internal reporting system with the exception of some noble gas releases that had minor dose consequence. A list of the radiological spills or releases to the environment that were identified in either the licensee's internal or external reports are summarized in the Chronological Listing of Events and NRC Response (Supplement B-1 to this report). This information references the NRC inspection reports(s) in response to specific incidents and denotes if an agency enforcement action was taken.

1.2 External Notifications and Reports

A review of all Haddam Neck Abnormal Occurrence Reports (AORs) from 1967 through 1975 and all Licensee Event Report (LERs) from 1976 through 1997 was conducted. Those reports pertaining to releases, spills, abnormal radiological control practices or radiation monitoring deficiencies are listed in the event summary (Supplement B-1). Most of the LERs were not associated with radioactive releases or spills of radioactive materials. Ten reports described on-site spills of liquid radioactivity that contaminated areas of the site inside the protected area fence. These contaminated areas were subsequently decontaminated. Twelve reports described unplanned offsite releases of gaseous radioactivity.

The LER notification system took effect in 1976. Prior to 1976, the licensee submitted AORs informing the NRC of events not in conformance with its Technical Specifications. In general, the quality of AORs and LERs varied until the LER reporting rule (10 CFR 50.73) became effective in 1984. The purpose of the rule was to eliminate reporting events of low safety significance and to require more thorough documentation and analyses of reported events. The rule requires the licensee to report to the NRC within 30 days after any event that meets the criteria.

Approximately 650 AORs or LERs have been written for Haddam Neck since the start of commercial operation in the late 1960s. From the start of the plant until the LER rule became effective in 1984, Haddam Neck wrote relatively fewer AORs and LERs than its industry peers on a per operating unit basis. The reports often lacked detailed root cause analyses and offered few follow-up actions. When the LER Rule came into effect in 1984, the overall quality of the reports improved. Better root cause determinations were provided and corrective actions were more inclusive. A detailed evaluation of the safety significance of a particular event is included in all LERs submitted after 1984 which means, for events involving releases of radioactive materials, that dose assessments to the public were evaluated. For the remainder of the 1980s, Haddam Neck submitted an average of 30 LERs each year which was comparable to the average of 28 issued by its peers in the industry. However, in 1986, 48 reports were submitted to the NRC, an abnormally high number compared to the industry average. In the 1990s, Haddam Neck had issued an average of 26 LERs each year, which is nearly twice the industry average of 14 LERs over the same time period.

Corrective actions were not always immediately effective nor timely to prevent recurrence. For example, there were four separate reports between June 1976 and February 1980 describing leakage from the radwaste discharge pipe which contaminated the sand beneath some on-site asphalt. The licensee's initial follow up was not entirely effective. Four years passed before the licensee ultimately corrected the problem. After each of these occurrences, the licensee excavated the contaminated sand and shipped it to an authorized radioactive burial facility.

Between March 1976 and September 1977, there were three instances of Waste Gas Decay Tank rupture disc actuation resulting in the release of 19 curies of noble gases to the environment. These disc actuations were a result of a field change to the design of the system when it was installed in 1975. A system modification was completed in 1977 shortly after the third failure. The modification reduced the problem but did not return the system to the original approved design. Although unplanned, these releases were monitored and did not exceed NRC off-site dose limits for members of the public.

A number of reports indicate repetitive occurrences of releases or spills of radioactive material. Of particular note are the following:

- Unplanned Noble Gas Releases

waste gas decay tank rupture disc actuation (11.6 Ci)	LER 76-08
waste gas decay tank rupture disc actuation (0.1 Ci)	LER 76-15
waste gas decay tank rupture disc actuation (7.4 Ci)	LER 77-06
degasifier rupture disc actuation (15.8 Ci)	LER 79-06
waste gas decay tank vent valve left open (19.7 Ci)	LER 85-25

- Leaks in Radwaste Discharge Pipe Contaminate Underground Sand

  LER 76-13, LER 77-0, LER 78-03, LER 80-07

Although these repetitive events indicate a potential weakness by the licensee to adequately identify root causes and appropriate corrective actions, none of the reported releases exceeded NRC dose limits to members of the public in unrestricted areas.

Through the review of licensee documentation, the team did not identify any releases where the licensee failed to notify the NRC as required by regulations. The number of LERs regarding inadvertent releases of radioactive material (22) is small when compared to the total number of LERs (650) and the total number of radiological incidents (125) identified by the licensee in the historical review of the site. However, in most cases where the licensee had a radiological incident, a notification was not required by NRC regulations in 10 CFR 50.72 or 10 CFR 50.73. Although the criteria for reporting events has changed with revisions to 10 CFR 50, the team's review indicated that the licensee properly reported the events in accordance with the regulatory requirements in effect at the time. For instance, in 1989, contaminated liquid was released to a leach field outside the protected area fence through the storm drain system from the Spent Fuel Building. The licensee classified this unmonitored radioactive liquid release as an Unusual Event in accordance with its Emergency Plan and made a one-hour telephone call to the NRC pursuant to the requirements of 10 CFR 50.72. Because the drain system at Haddam Neck is not considered a safety system and the release did not result in concentrations in unrestricted areas greater than 20 times the applicable concentration limits, written notification (LER) was not required under 10 CFR 50.73 and an LER was not issued. Although LERs were not required by NRC regulations for all radiological occurrences, the total radioactivity released was included in annual and semi-annual effluent reports as required for unmonitored or uncontrolled releases of radioactive material. Licensee effluent reports are reviewed by the NRC as part of the NRC core inspection program and the inspection reports document the acceptability of the licensee's effluent reports.

Conclusion

Haddam Neck reported an average number of licensee events during its commercial operating life when compared to the industry. Releases of radioactive material were reported as required. Reported releases did not exceed NRC limits to members of the public in unrestricted areas. Corrective actions were not always timely or effective to prevent recurrence. The team found that the licensee properly notified the NRC regarding radiological spills and inadvertent effluent releases.

2. NRC RESPONSE TO SIGNIFICANT EVENTS

Scope

The preliminary files for the 125 incidents that the licensee had identified in their Historical Site Assessment Data Table (Supplement A-1) were reviewed. Seventy incidents were evaluated by the team to determine the extent of NRC knowledge and response to these incidents (Supplement B-1). These 70 events were selected because there had been a notification to the NRC or the team believed the event could have an impact on the eventual decommissioning of the reactor site. The team reviewed approximately 600 inspection reports for the period from November 1967 through October 1997. The NRC Systematic Assessments of Licensee Performance (SALP) reports for Haddam Neck were reviewed to identify NRC awareness regarding overall licensee performance. The team also reviewed the NRC database regarding enforcement actions for Haddam Neck to determine what actions had been taken for the areas of interest within this report. In addition, the escalated enforcement history was reviewed for all other nuclear power reactor (Part 50) licensees in the United States to determine the type of enforcement taken in the areas of radiological effluents, radioactive material release, radioactive waste systems operation and fuel performance.

Details

2.1 General NRC Inspection and Event Follow-up Guidance

Routine NRC inspections were performed periodically since the start of operation at the Haddam Neck facility, with special inspections for significant events. In 1980, a resident inspector was assigned to each operating reactor site. The resident inspections increased the total inspection time at the site and covered a variety of areas, including plant operations, maintenance, engineering, radiation protection, security and follow-up to events. The resident inspection activities were supplemented by various specialist inspections in each major inspection area.

Specialist inspectors used various NRC Inspection Procedures (IPs) for guidance in a selected area of licensee activities (i.e., radiological controls). One major emphasis in NRC inspection procedures is a review of the effectiveness of the licensee's programs for radiation protection, radiological controls, radiological effluents, environmental monitoring, and radiological waste processing. There is limited guidance for radiological event response or NRC follow-up, including the review of the licensee's actions. The IPs used by a specialist inspector for a radiological event could include the following:

- IP 83750, titled "Occupational Radiation Exposure"
- IP 84750, titled "Radioactive Waste Treatment, and Effluent and Environmental Monitoring" and
- IP 83726, titled "Control of Radioactive Materials and Contamination, Surveys and Monitoring."

NRC IP 83726 provides guidance to inspectors in the area of radiological contamination controls. The current revision of the procedure for radioactive materials and contamination controls is very similar to the version used in the 1980s and puts emphasis on personal contaminations (skin or clothing), instruments and procedures for surveys and monitoring, proper clean-up of contamination (not specific to on-site or off-site) and reduction in volume of contaminated trash. The inspection procedure gives general guidance for reviewing the licensee's surveys, monitoring records and releases of potentially contaminated material to unrestricted areas. References include IE Bulletins, Circulars and Information Notices from 1980, 1981, 1985, and 1986. However, NRC Information Notice 88-22, "Disposal of Sludge from On-site Sewage Treatment Facilities at Nuclear Power Stations" is not referenced. This Information Notice discussed the need for licensees to perform radiation surveys of representative samples of materials under conditions that provide an LLD appropriate to measurements of environmental samples. NRC IP 83750 provides general guidance, similar to IP 83726, for inspectors in this topical area.

Effluent and environmental monitoring specialists use NRC IP 84750. This inspection procedure provides guidance for verifying dose commitments to the public from liquid and gaseous releases. With the exception of the verification on dose calculations, there is no other guidance on follow-up to a radiological event.

Based on the team's review of these NRC inspection procedures, there appears to be limited procedural guidance for inspectors to assess the adequacy of licensees' remediation efforts after a radiological event and the release of potentially contaminated volumetric materials from a licensed facility. Also, the responsibility for review of events involving licensed radioactive materials found outside the protected area is assigned on a case by case basis rather than being defined in NRC inspection guidance.

NRC inspections and event follow-up at Haddam Neck were performed in accordance with various inspection procedures. Inspection procedures provided general guidance for areas of review. Event reviews were principally focused on the consequences relative to applicable regulatory requirements regarding radioactive effluent or material released to the environment and personnel (worker) exposures. Other inspected areas, such as contamination events/decontamination efforts, which did not have a specific regulatory requirement, were reviewed using accepted industry practices and generic radiological controls standards.

Other guidance that was available to NRC inspectors is contained in various NRC regulatory guides. NRC Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures and Components Installed in Light-Water-Cooled Nuclear Power Plants," contains guidance on how to control the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid waste during normal reactor operation, including anticipated operational occurrences. The guidance states that all tanks located outside reactor containment and containing radioactive materials in liquids should be designed to prevent uncontrolled releases of radioactive materials due to spillage in buildings or from outdoor tanks. The guidance further states that all tank overflows, drains and sample lines should be routed to the liquid radwaste treatment system. Indoor tanks should have curbs or elevated thresholds with floor drains routed to the liquid radwaste treatment system. The design should include provisions to prevent leakage from entering unmonitored and nonradioactive systems and ductwork in the area.

NRC regulations require that power reactor licensees monitor radioactive effluent releases and ensure that the radiation exposure to a member of the public from the releases be as low as is reasonably achievable. Guidance on how to implement these requirements is contained in Regulatory Guide 1.109, Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I. The guide contains the basic calculational models and parameters for the estimation of radiation doses to man from significant radioactive effluent release pathways. The guide discusses that a pathway is considered significant if a conservative evaluation yields an additional dose increment equal to or more than 10 percent of the total from all pathways considered in Regulatory Guide 1.109. Therefore, the guidance states that monitoring is not required for liquid effluent pathways that would contribute less than 10% to the total annual exposure from all other liquid pathways. Also, an NRC health physics policy position (HPPOS-007) established in 1981 determined, based on the difficulties in monitoring radioactive discharge into storm sewer drains, the associated costs for installation and operation, general knowledge of past experiences with this particular type of unmonitored release from reactor operations and the small potential effect on public health, that requirements for monitoring storm sewer drains were unwarranted.

2.2 NRC Event Follow-up Inspection Reports

The review team developed a timeline that compares the events/incidents in NRC inspection reports and licensee reports (plant reports, LERS, etc.). The timeline is attached to this report as Supplement B-1. The time line indicates events that were reported by the licensee to the NRC and whether the events were documented by the NRC. It also indicates radiological events that resulted in NRC escalated enforcement. The inspection record indicates that the inspectors generally reviewed licensee event reports of spills or releases and associated licensee follow-up actions, worker contamination events, facility contamination status and decontamination efforts, and worker radiation exposures.

Typically, the licensee's documentation of unmonitored releases was reviewed as part of the routine NRC inspection of the environmental and effluent monitoring programs. To assess the significance of the event, inspectors reviewed documentation regarding the quantity of radioactive material released and the expected dose consequence from the release.

NRC inspectors started documenting the review of LERs for Haddam Neck around 1980. Until that time, very few AORs or LERs were documented in NRC inspection reports. The NRC staff generally documented most significant events or LERs that resulted in a potential release to the environment. However, there is not much detail written for most events. After 1990, a policy was implemented to track the review of all LERs so that each LER follow-up was identified in the inspection reports. This report requirement is included in the NRC Manual Chapter 0610, titled " Inspection Reports." The thoroughness of review and documentation regarding these events was dependent upon the significance of the LER and the extent of the inspector follow-up.

Although reported by the licensee to the NRC, some of the events at Haddam Neck were not addressed in NRC inspection reports. Even though the review team was unable to determine the reasons some events were not documented in NRC inspection reports, it was apparent that most events resulted in very low potential radiation exposures and had no safety consequence.

Based on the seventy selected events the following were not documented in NRC Reports:

- Improper waste discharge valve lineup in September 1977 caused 1400 gallons to go to the Spent Fuel Building Sump and then to the aerated drain tanks (radwaste system).
- High tritium background measurements in river water samples taken near the discharge canal in 1977 & 1978.
- Borated Water Storage Tank (BWST) heater leaked into the A & B heater wells and contaminated the in-house heating system (secondary system) in November 1978.
- Radioactive contamination found outside RCA but within the owner-controlled property in February 1981.
- 400 gallons of liquid radioactive waste (total activity of 3 microcuries) discharged from the radwaste test tank to the river due to valve mispositioning in December 1983.
- Various resin liner overflows in 1984 that resulted in local contamination on the site.
- Radiologically controlled area drains overflowed to the yard drain in March 1985.
- Broken drain line due to freezing on a temporary chemistry trailer in January 1986 that resulted in a small on-site contamination.
- Spill of component cooling water to a storm sewer in March 1990.

For the events that were documented, the review team found that larger spills/releases generally had thorough NRC review and follow-up regarding the licensee's corrective actions, but minor events had limited or no documented review. Most events were merely mentioned in inspection reports and no assessment was made of the licensee's corrective actions.

An evaluation of the inspection reports indicates that the NRC response to events at Haddam Neck varied. In particular, on-site contamination events often did not result in the use of a region-based specialist inspector. For example, trenching in the RCA in 1988 resulted in discovering contamination from a previous event in 1983. The licensee used a remediation level of 1E-4 microcuries/gram (μCi/g). Soil greater than this was treated as radioactive waste. However, it was not clear how the remaining soil was treated. Based on a review of the inspection reports, no specialist inspector evaluated the licensee's actions. In 1988, the licensee conducted an RCA refurbishment project. The NRC noted in the licensee's October 1988 Contamination Reduction Summary that many items within the RCA identified as waste had been either disposed of or released for unrestricted use. If the material was released, this value is a factor of 1000 higher than the existing NRC guidance (1 E-7 μCi/g) for an acceptable detection capability to detect the presence of license radioactive material prior to its release from a licensed site. Since no NRC follow-up item was identified, this issue was not re-examined by the NRC until the current site characterization activities.

Examples of NRC follow-up to significant events is outlined below:

- Degasifier event in 1979

  An event involving the release of steam and water to the plant stack was detailed in NRC Region I Inspection Report 50-213/79-21, dated February 20, 1980. Connecticut Yankee notified the Region I office at 4:45 PM on December 16, 1979, regarding an unplanned release earlier that day (5:45 AM) from the plant stack at Haddam Neck. The event happened during boron dilution operations and resulted in the venting of approximately 800 gallons of water/steam. The estimated radioactive release (approximately 16 Curies) of krypton and xenon lasted about 10 minutes. The Region I office dispatched three inspectors to the site on December 17, 1979, to review the licensee's response and corrective actions. The inspectors concluded the that there had been no release of radioactive liquids to the environment; the steam generator blowdown and containment particulate high radiation alarms were caused by high background radiation from the water in the exhaust duct; the release of particulate activity was less than 0.1% of the Technical Specification limits; the release of iodine activity was less than 15% of the Technical Specification limits; the release rate of noble gases exceeded the Technical Specification limits by a factor of 5.2 for a 10 minute period (non-compliance); and the potential serious radiological consequences of the event dictated a far more timely notification to the NRC than was actually made. The inspectors also raised concerns regarding the sampling for radioactivity in the stack. The calculated dose from this event to a member of the public at the site boundary was calculated to be less than 0.5 millirem. A Notice of Violation was issued to the licensee for exceeding the instantaneous release limit in Environmental Technical Specification.

Although the NRC response was timely and included a review of the licensee's actions, the inspectors did not document review of the adequacy of the stack monitoring relative to this event or examination of the licensee's area surveys after the event. The monitors were not designed for a vaporized steam release because vaporized steam was not typically released through the plant stack. In subsequent site radiological surveys performed by the licensee in 1980, numerous areas of localized radiological contamination were found outside the radiologically controlled area but within the owner-controlled area. Most areas of contamination were believed to be a result of this 1979 plant stack release. Additionally, once the licensee found the contaminated areas, the NRC staff documented the contamination in inspection reports but did not perform any further follow-up.

The team's review of the licensee's investigation report, dated April 1980, of this incident, identified that the licensee's dose assessment did not follow regulatory guidance. The dose assessment averaged the potential dose from particulate contamination over the skin of the whole body. The licensee's reported potential dose was 0.7 millirem. However, the dose when calculated over 1 square centimeter, which is consistent with regulatory guidance, was 6.3 rem. Based on the above, this event had the potential to result in doses approaching the occupational quarterly limit for the skin of the whole body. (The limit in effect in 1979 was 7.5 rem/quarter.)

- Multiple underground leaks of contaminated systems

From 1976 through 1980, there were several occurrences of leaks from underground pipes (steam generator blowdown, service water, and liquid waste test tank) that contributed to the tritium concentration in the local groundwater. The NRC response to these events included a review of the licensee's corrective actions. The NRC follow-up to these events and more recent follow-up to the licensee's actions for a leak in the RWST were documented in NRC Inspection Report 50-213/97-11. The results of the NRC review of the licensee's data indicated that the levels did not exceed the EPA levels for tritium in drinking water.

- Degradation of fuel cladding in 1979, 1989 and 1991

The NRC responded to the 1979 fuel cladding defects with issuance of License Amendment No. 52, which authorized changes in the fuel design and reduced the allowable peak power level in fuel rods. The safety evaluation issued with the amendment included consideration of fuel performance for the three years following the 1979 event and concluded that the changes effectively reduced the number of defects to few, if any, leaking rods.

The NRC responded to the 1989 fuel defect event with issuance of License Amendment No. 134, which limited the number of leaking fuel rods to no more than 160. It also imposed surveillance requirements to assure that throughwall fuel cladding penetrations would be promptly detected and accurately quantified. The licensee established a Fuel Recovery Program to respond to the event and sent the results of that program, such as proposed changes to the license conditions and analysis of fuel inspection findings, to the NRC.

The NRC responded to the 1991 fuel defect event with increased inspector attention to the fuel integrity monitoring program. The inspector noted that the licensee had discussed fuel integrity and the application of the associated seven-day LCO with NRC Headquarters after an indicated increase in the number of throughwall defects exceeded 160 rods following a power maneuver. However, the NRC did not perform a formal assessment of the licensee's 50.59 evaluation of the change made to the fuel monitoring program. A subsequent NRC inspection noted that preliminary results of the licensee's fuel inspection found 100 leaking rods. A follow-up report by the licensee summarizing its Fuel Recovery Program estimated 102 fuel rods had throughwall defects by the end of Cycle 16. (see Appendix A, Section 7 of this report).

- Discharge through the switchyard trench in 1989

Another example of NRC follow-up to an event was after the licensee inadvertently emptied radioactive liquid down an uncontrolled floor drain in the Spent Fuel Building during radwaste processing activities in January 1989. The drain was not labeled and the liquid discharged directly to an open trench that drained to a marshy area of the site. Radioactive material could then migrate into the discharge canal. However, freezing conditions reduced the amount of liquid that left the protected area. The licensee identified the radioactive material in February 1989 during a routine radiological survey of the site. The NRC reviewed the licensee's corrective actions and documented the actions in an inspection report (Reference NRC Inspection Report No. 50-213/89-02). However, the inspectors did not document any assessment of the licensee's corrective actions or perform independent measurements. The inspectors concluded, based on the licensee's conservative assumption that all radioactive material was released to the environment, that the resulting dose would be a small fraction of the whole body and maximum organ dose limits.

In the inspection report (50-213/89-02) cover letter, the NRC stated in part:

> *"We are very concerned that an unmonitored radiological release path has existed through Spent Fuel Building drains and that radioactive liquid entered these drains on at least one occasion. The issue of unmonitored release paths was brought to your attention in IE Bulletin 80-10. This area warrants your further consideration."*

No other NRC response or follow-up was documented at the time for this occurrence. However, this issue was reviewed by NRC inspectors in 1997. The licensee subsequently started a comprehensive review associated with IE Bulletin 80-10 (see Appendix A, Section 6 of this report).

- Reactor cavity seal failure in 1984

  The licensee had an event involving the reactor cavity seal failure in 1984 that resulted in 200,000 gallons of water discharged to the lower level of the containment building. Although there was no release to the environment, the lower level of the containment building was contaminated. The NRC reviewed the licensee's actions in response to this event in NRC Inspection Report Nos. 50-213/84-23 and 50-213/85-17. Escalated enforcement actions were taken by the NRC (EA 84-115) due to the operational aspects of this event.

Few independent measurements were documented as performed by NRC inspectors after a spill/release/event. Typically, the NRC inspectors reviewed the licensee's program for analyzing environmental and effluent samples but did not perform actual measurements. This was consistent with the NRC policy to review licensee programs instead of verifying individual measurements. However, the NRC Region I office employed a mobile laboratory that periodically visited power reactor sites and performed independent measurements as a check on licensee performance. Contamination control assessments generally relied upon the licensees' dose rate and surface contamination results. However, an independent verification, including soil samples, was performed by a specialist inspector in 1982 after alleged improper control of radioactive material on the licensee's site. The inspector concluded that the licensee's actions for remediation of contamination were complete and appeared adequate. The licensee's actions included surveying the areas, removing the contaminated soil and paving the area (NRC IR 50-213/83-02).

The NRC response and involvement from specialist inspectors has changed over the years. NRC event follow-up typically includes oversight of the licensee's corrective actions, review of dose assessments and calculations, and independent measurements for significant radioactive material releases. Recent NRC inspections performed in 1997 at Haddam Neck have provided oversight and independent measurements of the licensee's characterization surveys of the plant site and offsite areas.

2.3 NRC Enforcement Actions

The NRC enforcement history for Haddam Neck shows that enforcement actions were taken through the entire period of plant operation, continuing into the shutdown phase. Few enforcement actions were taken for spills or releases since the safety significance was very low.

The NRC enforcement history at Haddam Neck is typically representative of low-severity level violations for specific events. Radiological releases were seldom cited for escalated enforcement, because the amount of radioactivity released did not meet the NRC enforcement criteria for escalated enforcement. The releases of noble gas in excess of the Technical Specification limits were one example where the NRC cited numerous violations within a short time period (less than two years). One escalated enforcement action in 1979 was issued due to a breakdown in the radiological controls program that led to numerous violations of NRC regulations. The licensee developed a corrective action plan to upgrade the quality of the radiological controls program. After implementation of these corrective actions, NRC inspections reflected some improvements in the program.

However, several other violations were cited in the years following the implementation of the corrective actions.

The NRC team identified various plant events or conditions that potentially affected radioactive material and radiological controls over the operating period of the facility and must be considered during the licensee's site characterization effort. It is possible that some of these events or conditions are potential violations of regulatory requirements and were missed opportunities for the agency to consider and apply enforcement sanctions. These included the following:

- a modification of the radioactive waste process system in 1975, which was not adequately evaluated by the licensee in accordance with 10 CFR 50.59, resulted in radiological releases from the waste gas decay tank that were larger than they would have been if the system had been installed as originally designed and approved;

- local onsite contamination associated with various radioactive liquid waste processing activities that occurred in the open environment, though described in the design basis and covered by procedures and the Process Control Program, resulted in the migration of radioactivity to areas outside of the RCA that were not regularly surveyed or recognized as being potentially contaminated;

- systems designed as non-contaminated were continuously used after being contaminated without supporting safety evaluations and without implementing a periodic monitoring program until 1997;

- the conduct of radioactive waste handling activities in the spent fuel building in 1989 (an activity that appears to have been outside of the design basis) led to the release of radioactive materials to areas outside of the RCA, through an unmonitored and uncontrolled drain system;

- fuel cladding defects in 1979, 1989 and 1991 which increased the source term and radiation exposure hazards in the facility and resulted in the deposition of transuranic activity in many plant systems (radiological and non-radiological), consequently affecting the classification of certain radiological solid wastes from the site.

Regarding fuel cladding defects, the NRC team also determined that:

- defects in 1989 resulted in the plant exceeding a design basis limit (1% failed fuel assumed in the waste gas decay tank rupture accident), but it was not recognized or reported as such; and

- defects in 1991 contributed to the licensee modification of the fuel monitoring program, used to implement the surveillance requirements of the technical specifications, which may have been an unreviewed safety question and an unapproved change to the technical specifications.

While some of these conditions may be potential violations of agency requirements, the doses to workers and the public resulting from these situations were within the requirements of 10 CFR 20. The apparent safety significance and dose consequence to plant workers or to members of the public were low in these instances based on the NRC review of the licensee's environmental monitoring, radiological effluents, and radiation protection programs. These potential violations will be further reviewed by the NRC staff and considered for future enforcement actions in accordance with the NRC Enforcement Policy.

NRC Region I imposed a Civil Penalty of $650,000 in May 1997 for numerous violations at Haddam Neck regarding design errors during design changes, making facility changes without performing adequate safety evaluations, inadequate procedures and failure to follow procedures. NRC follow-up on the licensee's corrective actions in response to the May 1997 escalated enforcement action is continuing.

NRC enforcement was taken for a few radiological contamination events at Haddam Neck, in particular recurring events. The NRC inspectors generally documented the corrective actions taken for each violation in subsequent inspection reports. The violations were tracked as open items until the corrective actions were reviewed by NRC inspectors and verified to be implemented by the licensee. The inspectors generally found the corrective actions to be appropriate for these events. Notwithstanding this regulatory documentation process, the adequacy of the corrective actions in subsequent years was not generally revisited.

Escalated enforcement at other Part 50 licensed sites (nuclear power reactors) was not very common for the areas of radwaste design/operation, failed fuel events, radiological releases/contamination events or effluents/environmental monitoring problems. The review team believes this is consistent with the overall operation of the facility and other circumstances that are used to determine if escalated enforcement is necessary (i.e., effectiveness and timeliness of corrective actions, recurrence of events, self-identification and safety significance).

The first documented enforcement action for radiological activities at another facility was a $5,000 civil penalty (part of a larger $19,000 civil penalty) levied against Consumer's Power Company at the Palisades Plant in 1974. The enforcement action was taken relative to a discharge of gaseous waste to the plant stack, which resulted in a release to the environment. This was a Severity Level II violation.

Another enforcement action in 1974 was taken for an unmonitored release to the environment of laundry wastes at the Dresden Station. Commonwealth Edison was fined $16,000 for this violation as part of a larger $25,500 civil penalty.

Duke Power was fined $21,500 for six infractions and one deficiency when the operators at the Oconee Plant released 3 Curies of Iodine-131 into the river in 1977.

There were two escalated enforcement actions taken for violations at the Brunswick Station in 1980. Carolina Power and Light Company was fined $24,000 for a violation involving the operation of the auxiliary boiler system while it was contaminated and no safety evaluation was performed to determine the potential unmonitored release of radioactivity from the system. The licensee was fined $89,000 for the disposal of licensed material without authorization from the NRC and release of contaminated materials for unrestricted use. This event prompted the issuance of IE Bulletin 80-10.

The operators of the Hatch Plant were fined $95,000 in 1981 for violations involving high radiation levels in an unrestricted area and release of waste oil with unknown radioactivity.

In 1983, Public Service Electric & Gas Company was fined $20,000 for a violation involving the containment gaseous, particulate and iodine monitor at the Salem Nuclear Power Plant. The monitor sample line was capped and the monitor was out of service for several days.

Sacramento Municipal Utilities District was fined $100,000 for an unplanned release from the Rancho Seco Nuclear Power Station in 1986. A matching fine was issued to the operators of the San Onofre Plant (Southern California Edison) for various violations, including the release of radioactive materials to an unrestricted area. In 1991, the Power Authority of the State of New York was fined $137,500 for violations, including an unplanned release of radioactive materials to an unrestricted area.

More recently, Public Service Electric & Gas Company was given various violations (Severity Level III) for an operational event in 1996 that released 25 gallons of steam and water to the environment from the liquid radwaste system at the Hope Creek Nuclear Power Plant. The release contaminated buildings, personnel and vehicles on the site. At least one contaminated vehicle was removed from the site without a radiological survey. The total release was estimated to contain approximately 85 millicuries. GPU Nuclear Corporation was charged with five violations (highest Severity Level of IV) for events which included an unplanned release due to a human performance error in 1997. Water from a radioactively contaminated system was used to flush another clean system component, which resulted in a minor release of radioactivity to the environment. This was not the first unplanned release to the environment at the facility in a short period of time, and even though the potential doses to the public were well below the regulatory limits, the NRC issued the violation to express the concern regarding repeated problems.

Note: The increased civil penalty values that have occurred over time are most often due to the changes in the NRC enforcement policy that have increased the base civil penalty for escalated enforcement.

Conclusions

NRC follow-up to radiological events at Haddam Neck varied based on safety and operational significance. The major radiological events were documented in inspection reports. None of these cases resulted in exposure to the public in excess of the annual regulatory limits specified in 10 CFR 20. However, events in 1979 that resulted in approximately 40 discrete areas of the site with fission product activity could have resulted in skin contamination with doses near the quarterly occupational limits in 10 CFR Part 20. Review of licensee's programs was the primary method used to determine the adequacy of licensee data, but some independent measurements were performed to verify that the licensee's programs and processes were accurate. Response to significant events has improved in the 1990s and specialist inspectors are more likely to be involved in the NRC follow-up and evaluation of the licensee's corrective actions to prevent recurrence. NRC inspection policy and procedures do not provide criteria for NRC follow-up response to radiological events that have limited safety significance. The amount of follow-up is at the discretion and direction of NRC management and is usually dependent upon inspection priorities and safety significance.

Enforcement actions at the Haddam Neck plant were generally consistent with the existing policy of the NRC and practices that evolved over time. Enforcement was taken for some radiological events, including recurring problems, however, the enforcement actions were low level and not escalated. This is typical for events at other nuclear power reactors, but escalated enforcement action was taken in some cases at other power reactors in the past. These cases must be reviewed in detail to determine the enforcement criteria that was applied and the civil penalties that were imposed upon the operators of the facilities. Escalated enforcement has been more common since 1980. Additionally, escalated enforcement is usually taken when an operational error or a failure to follow procedures is also cited with an unplanned radioactive release.

The NRC Historical Review Team noted a number of past licensee actions that may not have been in compliance with NRC requirements existing at the time. These items will be further reviewed by the NRC staff to determine if enforcement actions are appropriate. This review will consider the relationship of the issues to the current licensed activities at the site and the need for corrective actions to prevent recurrence.

Table B
Enforcement Actions for Areas of Concern

Year	Enforcement Action	Severity	Description
1969	Non-compliances (2)	N/A	Release of radioactive isotopes to an unrestricted area (200 gallons to the discharge canal); 700 millirem/hour radiation field at fence line
1970	Non-compliances (2)	N/A	Inadequate surveys; stack recorder not inking
1971	Non-compliance	N/A	Restricted area dose rates above limit of 100 millirem in a 7 day period
1973	Non-compliance	N/A	Monthly tests of containment air filtration system not performed
1975	Infractions (2)	N/A	Failure to survey in containment
1976	Deficiencies (2)	N/A	Failure to perform a gamma isotopic analysis of a weekly gaseous particulate filter sample
1979	Infractions (9); $27,500 CP management mtg.	Escalated	Numerous violations/weaknesses in radiological controls program and release through the plant stack
1980	Infractions (3)	N/A	Various unplanned noble gas releases in excess of Tech. Spec. limits
1980	Violations (3)	Level V	Environmental monitoring program deficiencies
1981	Violation	Level IV	Radioactive material transferred as scrap to an non-licensed individual
1984	Violation	Level IV	Improper radwaste processing
1984	Escalated; $80,000 CP	Level II	Cavity seal failure, release of contaminated water to the containment floor
1984	Escalated; No CP	Level III	Radiation dose for worker exceeded regulatory limits
1986	Violation	Level IV	Improper use of radwaste compactor spread contamination to outdoor areas
1986	Violations (8)	Level IV	Radwaste deficiencies
1986	Escalated; $50,000 CP	Level III	Radiation dose for worker exceeded regulatory limits
1995	Escalated	Level III	RHR system
1996	Escalated (Pending)		Unplanned exposures to workers in the fuel transfer canal
1997	Escalated, $650,000 CP	Level II, III, IV	Design control issues
1997	Escalated; No CP	Level III	Corrective action problems, safety evaluations

Chronology of Events and NRC Responses (SUPPLEMENT B)

Licensee Identified Event	YEAR	NRC Regulatory Response/Followup
Spill in Boron Recovery Area due to broken pipe (5/6/69) AO 69-07 6/19/69	1969	Non-compliance Technical Specification (TS) 4.6, evaluation of release adequate IR 69-02 & IR 69-03 6/16/69 10/29/69
Storage drum 700 mrem/hr at the fence AO 6/30/69		Non-compliance 10 CFR 20.201 Non-compliance 10 CFR 20.20.105(b) IR 70-04 & IR 71-01 7/8/70 2/11/71
Iodine released due to operator error with penetration seals (4/18/71) AO 5/10/71	1971	Non-compliance 10 CFR 50.59 Mod. to VCT without safety evaluation IR 71-02 6/9/71
Unplanned airborne release from demineralizer due to operator error (5/19/72) AO 72-02 6/23/72	1972	Commitment to evaluate potential unmonitored pathways IR 72-03 5/24/72
Unplanned airborne release due to leak in purification system valve (6/21/73) AO 73-06 6/22/73	1973	Event reported to AEC on 7/3/73 Inspection Follow-up Item (IFI) IR 73-05 9/21/73

AO = Abnormal Occurrence
ACR = Adverse Condition Report
LER = Licensee Event Report

PNO = Preliminary Notification of Occurrence
IR = Inspection Report
EA = Enforcement Action
UE = Unusual Event

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Chronology of Events and NRC Responses

Licensee Identified Event	YEAR	NRC Regulatory Response/Followup
Unplanned liquid release from letdown system due to procedural error (6/21/73) AO 73-07 7/3/73	1973	Event reported to AEC on 7/3/73 Inspection reviewed Event IR 73-05 9/21/73
Radioactive Waste Storage Tank heater valve leaking Storm drain contaminated AO 73-11 11/2/73		Corrective action addressed periodic sampling of drains IR 73-09 2/1/74
Unplanned gaseous release from volume control tank detected on 4/26/74 AO 74-7 4/29/74	1974	Report noted plans to add tritium monitoring to main vent IR 74-11 12/18/74
Waste Gas Decay Tank Rupture Disc failed LER 76-8/36 4/29/76	1976	
Waste Gas Decay Tank diaphragm rupture PIR 76-79 6/29/76		

Chronology of Events and NRC Responses

Licensee Identified Event	YEAR	NRC Regulatory Response/Followup
SG blowdown waste discharge pipe and service water effluent line leaked below PAB floor Tritium released underground LER 76-13/990 7/12/76	1976	Unresolved item (URI) - Contamination same as discharge test tank IR 76-11 7/13/76
Waste gas decay tank rupture disc failed Design change to relocate disc LER 76-15/3L 7/27/76 PDCR 237 3/3/77		
RWST heater dripping water on blacktop PIR 76-140 12/20/76		
Increased H-3 from leak of steam generator blowdown line and service water line LER 77-1/3L 1/14/77	1977	Releases reported in Semi-annual Effluent Report 1/1/77 - 6/30/77
Leakage of 1000 gallons from recycle test tank to diked area LER 77-2/3L 3/10/77		IFI - No release to environment outside of radiation control area (RCA) IR 77-05 3/25/77

Chronology of Events and NRC Responses

Licensee Identified Event	YEAR	NRC Regulatory Response/Followup
Improper valve line-up for processed waste to Spent Fuel Pool Bldg sump and then to Aerated Drain Tank PIR 77-83 9/6/77	1977	
Waste Gas Decay Tank disc rupture Instantaneous release rate exceeded LER 77-06PE 9/26/77		Unplanned release of noble gas in excess of Env. T.S. limit IR 77-20 & IR 78-03 11/30/77 5/2/78
Tritium activity of river water near discharge canal exceeded control station LER 77-07E 11/4/77		NRC recommended continuous or short-interval composite sampler IR 78-03 5/2/78
Effluent rad monitor out of service Liquid waste batch sampled LER 77-08PE 11/9/77		
Third leak steam generator blowdown and service water discharge LER 78-03/3L 3/31/78	1978	URI - Repairs planned for next refuel outage IR 76-14-10 IR 78-14 11/13/78 6/28/78

Chronology of Events and NRC Responses

Licensee Identified Event	YEAR	NRC Regulatory Response/Followup
Tritium activity in river water near discharge canal exceeded control station LER 78-10E 10/27/78	1978	Released concentrations were below liquid effluent limits IR 80-25 3/27/81
Boron Water Storage Tank heater leaking into A&B wells PIR 78-120 11/25/78		
Contamination of RCA yard area around BWST diked areas. PIR 78-122 11/29/78		
Abnormal degradation of fuel cladding in Batch 8 fuel assemblies LER 79-01 2/24/79	1979	Fuel cladding degradation identified 13 non-compliances in HP program IR 79-05 & IR 79-06 3/13/79 7/31/79
Total activity in RWST greater than TS limits (2/19/79) LER 79-003E 3/19/79		Corrective action included new resin-bed in service IR 79-05 IR 79-06 IR 79-07 3/13/79 7/31/79 8/21/79

Chronology of Events and NRC Responses

Licensee Identified Event	YEAR	NRC Regulatory Response/Followup
Liquid waste discharge line leaked contaminating soil near hot machine shop driveway PIR 79-92 8/10/79	1979	Identified issues as poor practices in response to spill IR 79-07 8/21/79
Unplanned radioactive gas release from degasifier to the environment 12/16/79 LER 79-006E 12/17/79		Non-compliance of TS release rate, roof sealed with asphalt, drain cleaned, 12,000 gallons flushed IR 79-21 2/20/80
Leak of Boron Waste Storage Tank (BWST) to diked area LER 80-005 2/7/80	1980	
Fourth leak where S/G blowdown line connects to the service water discharge line LER 80-007 2/27/80		
Drain line from diked areas to the storm drain broken MSM-35-80 4/28/80		

Chronology of Events and NRC Responses

Licensee Identified Event	YEAR	NRC Regulatory Response/Followup
Contamination outside RCA on 3/10/80 Licensee Report Licensee Phone Call 3/10/80	1980	Follow-up to licensee's telephone notification IR 80-04 6/25/80
Three unplanned releases of noble gases during slucing of spent resins to the aerated drain tank on 5/3/80 & 5/4/80 Licensee ENS Calls 5/3/80 & 5/4/80		Non-compliance TS 2.4.3.1 violation closed 4/81 IR 80-07 IR 80-09 IR 81-05 9/2/80 6/2/80 8/81
VCT vented to plant stack Licensee ENS Call 5/28/80		URI to review OP procedures IR 80-07 8/27/80
Aerated drains tank waste evaporator spill LER 80-015 10/20/80		
Radioactive contamination outside of RCA PIR 81-10 2/9/81	1981	

Chronology of Events and NRC Responses

Licensee Identified Event	YEAR	NRC Regulatory Response/Followup
Primary to secondary S/G leak LER 81-09 8/20/81	1981	Commitment to review non-radioactive waste systems by 11/30/82 noted SALP (Cycle 3) 7/2/82
Auxiliary Building exhaust duct to main stack developed crack in weld seam LER 81-015 9/17/81		1% of exhaust flow bypassed the stack. Soil contamination identified in 1982 IR 81-12 11/23/81, IR 82-08-01 1/18/83, 83-02 2/22/83
Primary to secondary S/G leak (4/14/83) LER 83-008 5/12/83	1983	LER Follow-up for accuracy & corrective action IR 83-20 12/2/83
Radioactive water released to the discharge canal due to valve mispositioning PIR 83-137 12/13/83		
Resin liner overflow PIR 84-181 9/11/84	1984	

Chronology of Events and NRC Responses

Licensee Identified Event	YEAR	NRC Regulatory Response/Followup
Resin liner overflow PIR 84-182 9/13/84	1984	
Drained Refuel Pool to containment UE 8/24/84; LER 84-013 9/21/84		Design Change Deficiencies $80,000 Civil Penalty & Order IR 84-23 9/14/84; EA 84-115 12/13/84; IR 85-17 12/26/85
RCA yard drain overflowed to yard drain PIR 85-52 3/25/85	1985	
Design change in 1980 containment isolation valves may provide direct release pathway LER 85-017 & LER 85-012. Rev.2 8/14/85		Administrative controls to prohibit alternate letdown mode. IR 86-06 5/30/86
Unplanned release from waste gas decay tank 9/19/85 & 12/11/85 LER 85-025 10/18/85		URI - Operation of Waste Gas System IR 85-25 1/14/86 & IR 86-03 5/5/86

Chronology of Events and NRC Responses

Licensee Identified Event	YEAR	NRC Regulatory Response/Followup
Broken drain line from temporary chemistry trailer near waste gas building PIR 86-23 1/22/86	1986	
Dredging of discharge canal 11/13/86		Environmental & Effluents Inspections did not identify removal of sediment
Ground water leakage into containment cable vault LER 87-007-00 7/7/87	1987	
Drain Hose Spill Memo 7151 8/2/87		Hot Particle Program 70 particles identified by licensee IR 87-21 9/11/87
Failure to take service water effluent samples LER 88-014-00 6/7/88	1988	Violation was identified for same in 1986 IR 86-15 6/30/86

Chronology of Events and NRC Responses

Licensee Identified Event	YEAR	NRC Regulatory Response/Followup
Contaminated soil found while digging near manhole #11 inside RCA PIR 88-181 10/11/88	1988	Soil under sump not removed (IFI) IR 88-19 12/2/88
Highly contaminated water drained into uncontrolled drain that emptied into the 115 Kv yard trench PIR 89-35 2/24/89 UE 2/24/89	1989	Unmonitored release path<TS limits PNO 89-013 2/24/89 IR 89-02 5/17/89
Significant fuel damage identified on 11/17/89 LER 89-020-00 12/15/89		Fuel damage identified 9/25. UT of all fuel done 10/19. 122 fuel assemblies IR 89-16 11/9/89 IR 90-02 3/14/90
Spill of component cooling water to storm sewer PIR 90-52 3/22/90	1990	
Hoses dropped and contaminated an area of RCA yard PIR 90-65 4/11/90		

Chronology of Events and NRC Responses

Licensee Identified Event	YEAR	NRC Regulatory Response/Followup
Leak from RWST PIR 90-239 9/14/90	1990	RWST leak 5 to 50 gpd contained in Dike Sealed yard drains near tank IR 90-15 12/7/90 IR 90-19 1/7/91 IR 94-17 9/14/91
Sampling frequency not met LER 91-021-00 5/9/91	1991	
Spill from RCS to pipe trench PIR 91-149 8/12/91		Inspection noted corrective actions acceptable IR 91-16 9/20/91
Potential for radiological release Post LOCA sump recirculation LER 94-007-00 4/5/94	1994	Routine follow-up inspection IR 94-09 6/24/94
PAB supply heat exchanger drain line PIR 94-76 4/15/94		

Chronology of Events and NRC Responses

Licensee Identified Event	YEAR	NRC Regulatory Response/Followup
Gauges with fixed contamination released ACR 94-179 11/23/94	1994	Non-cited violation for contamination control IR 94-27 2/15/95
Contamination found in yard drain 4 during routine sampling PIR 95-067 2/12/95	1995	Short term corrective actions acceptable IR 95-08 3/25/95
Flooding in diesel rooms to discharge canal LER 95-002-00 2/22/95		Minor event with respect to contamination IR 95-06 2/24/95
Contamination found outside RCA PIR 95-250 8/1/95		Licensee identified & corrected IR 95-19 9/19/95
Contaminated hose to hydrolaze the tracks outside the SFB rollup door PIR 95-472 11/15/95		Contaminated hose to hydrolaze the tracks outside the SFB rollup door IR 95-27 1/25/96

Chronology of Events and NRC Responses

Licensee Identified Event	YEAR	NRC Regulatory Response/Followup
Tritium found in yard drains 4, 5, and 6 on 10/4/96 ACR 96-1185 10/7/96	1996	Tritium below reporting level IR 97-01 5/18/97
Calibration of RMS effluent monitors potentially inadequate LER 97-05 4/1/97	1997	Breakdown in calibration program IR 97-02 IR 97-03 3/21/97 7/7/97
Liquid effluent monitor inoperable due to low sensitivity LER 97-06 4/9/97		Inspection noted corrective actions acceptable IR 97-03 7/7/97
Radioactivity in sand near RWST ACR 97-0670 8/22/97		
Radioactivity found in closed loop cooling water system ACR 97-0694 8/29/97		URI - Licensee evaluation ongoing IR 97-07 8/18/97

Chronology of Events and NRC Responses

Licensee Identified Event	YEAR	NRC Regulatory Response/Followup
Radioactivity found outside the RCA at the shooting range ACR 97-0785 9/24/97	1997	Non-compliance cited. Follow-up Inspection activities continuing. IR 97-07 8/18/97 IR 97-08 10/30/97

APPENDIX C

BACKGROUND AND GENERAL REGULATORY PERSPECTIVE

A. BACKGROUND

Since November 1996, a series of radiological control performance issues have emerged at Connecticut Yankee Atomic Power Company's (CY) Haddam Neck Plant. An NRC inspection (50-213-96-12) conducted in November 1996, to review an event involving personnel exposure at Haddam Neck, revealed significant deficiencies in the licensee's Radiation Protection Program and its implementation. On December 5, 1996, CY announced the permanent shutdown of Haddam Neck[1] and indicated its plan to decommission the facility, an activity that would involve significant radiological work. Subsequent NRC inspections revealed continuing problems in the area of radiological control, including control of radioactive materials and maintenance of radiation monitoring systems.

Based on problems identified in these inspections, NRC determined that CY's apparent deficient performance and ability in radiation protection warranted immediate and comprehensive assessment and corrective action. Accordingly, NRC issued a Confirmatory Action Letter (CAL) on March 4, 1997, that confirmed the licensee's commitment to make improvements in its radiation protection program. In subsequent correspondence, the licensee committed to limit radiological work activities until radiation program improvement was accomplished.

As described by the CAL, the licensee conducted a comprehensive review and assessment of the radiation protection program, which revealed significant programmatic deficiencies. Accordingly, CY established a Radiation Protection Program Improvement Plan designed to significantly improve overall performance relative to: monitoring and control of radioactive material and contamination; radiological effluent monitoring and control; radioactive waste processing and handling; and control and monitoring of radiological work and radiation exposure of personnel. This licensee improvement effort is still in progress and is being closely monitored by the NRC.

In conjunction with radiation protection program improvement initiatives, CY initiated efforts to scope the radiological status of the facility and its environs. The purpose of this effort was to estimate the extent of on-site contamination (in normally radiologically controlled areas and on adjoining CY controlled property) that would require remediation to support decommissioning. As a result, CY discovered that licensed materials (i.e., contaminated soil, debris, construction materials and other articles) may have been improperly monitored and released for unrestricted use during the 30-year operating life of the facility. Consequently, CY expanded the characterization activity to establish plans to review previous practices relative to monitoring, control and release of the suspected materials; to determine possible or probable locations of such material; and to achieve retrieval or remediation, as necessary.

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[1]Connecticut Yankee Atomic Power Company correspondence to NRC, dated December 5, 1996.

Subsequently, CY confirmed the presence of licensed materials in various off-site locations, including private residences. The instances to-date involved only low-level or trace concentrations. Consequently, while detection of the material in unrestricted locations was unexpected, in all cases examined to date there was no apparent impact on public health and safety. However, the finding resulted in significant public concern about the past operation of the facility and possible impact on public safety. Currently, the NRC continues to monitor and evaluate CY's actions to identify and remediate off-site locations, maintain communication with the Connecticut Department of Environmental Protection (CT-DEP) and perform confirmatory sampling and analysis of suspect materials.

Additionally, the Federal Energy Regulatory Commission (FERC)[2] initiated a rate case hearing process in early 1997 to establish the validity of costs that may be applied to the rate base as a result of decommissioning of Haddam Neck. Testimony, provided on behalf of the Connecticut Department of Public Utility Control (PUC), identified several events, based on licensee records and information, which resulted in radiological soil and ground contamination of the facility over its 30-year operational life, that may have affected decommissioning costs. While the PUC testimony was focussed on CY's management and control, the statements and characterizations elicited strong public concern about the status of current public health and safety in the immediate vicinity, as well as the quality of licensee performance and regulatory effectiveness over the last 30 years of operation.

A number of radiological control performance issues emerged after the license announced plans to permanently shutdown. Multiple NRC-identified radiation protection program deficiencies; detection of contaminated materials in various off-site locations; and published testimony before the FERC engendered questions relative to the present extent of residual facility contamination and the ability of the licensee to adequately characterize the site for eventual site decommissioning. These matters also produced interest relative to the circumstances that resulted in on-site contamination and the detection of radioactive contamination in some off-site locations and the effectiveness of NRC's regulatory oversight.

These matters drew considerable attention from the public. The concern was shared by local and state government officials, including the Governor, the state's Attorney General and interested Members of Congress. Accordingly, on October 23, 1997, the NRC established an action plan for performing a limited historical review of radiological controls and area contamination issues at Haddam Neck.[3] This historical review was conducted by members of the NRC staff having expertise in licensing, inspection, and various aspects of radiation protection and reactor plant decommissioning. Primarily, the effort involved

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[2]In accordance with the Federal Power Act and the Public Utilities Regulatory Policies Act, the Federal Energy Regulatory Commission approves rates for the wholesale of electricity and transmission in interstate commerce involving private utilities, power markets, power pools, power exchanges and independent operators.

[3]Memorandum to L. Joseph Callan, Executive Director of Operations, from Hubert J. Miller and Samuel J. Collins, dated October 23, 1997, Action Plan Related To Radiological Control and Area Contamination Issues at Haddam Neck.

review of available licensee historical records and pertinent NRC regulatory documentation relative to the specified objectives of the action plan.

B. GENERAL REGULATORY PERSPECTIVE

The Nuclear Regulatory Commission's system of regulation is based on the fact that the primary responsibility for the safe design, construction and operation of any commercial nuclear power plant principally rests with the licensee. The NRC's primary function includes setting regulatory standards and specifications for radiological and nuclear safety relative to the conduct of licensed activities, ensuring compliance through inspection and enforcement, and conducting systematic assessment of performance. In this manner, the NRC and its licensees share a common responsibility to protect the public health and safety and the environment.

To accomplish this objective, the agency has established a system of licensing and regulatory activities that includes, among other functions, a formalized process for licensing and inspecting the operation of commercial nuclear reactors; the development and implementation of rules and regulations that govern licensed nuclear activities; investigation of safety-significant events or allegations of impropriety involving NRC-licensed activities; enforcement of NRC regulations and license conditions; establishment of working relationships with affected states; collection, evaluation and dissemination of information pertaining to operational safety of commercial nuclear power plants; and the audit of licensee performance and conformance with regulatory requirements, including radiological control and radiation protection.

The NRC's inspection role is accomplished by examining various aspects of licensee performance of activities relative to regulatory requirements. Periodic audits are conducted of licensee programs and processes that are necessary for the safe conduct of licensed activities to ensure they are established, implemented and maintained in accordance with the design and licensing bases. Relative to radiological aspects, the agency's normal inspection process is accomplished by periodically auditing the licensee's radiation protection program performance, including: quality of commitment to safety; technical capabilities relative to radiological monitoring , assessment and analyses; staffing, relative to selection, qualification and training of personnel; quality of processes and procedures; problem resolution and corrective action effectiveness; conformance with regulatory requirements and specifications; and the quality of efforts to maintain exposures to workers, the public and the environment as low as reasonably achievable (ALARA). Reactive inspection activities are usually conducted for emergent or abnormal conditions that have the potential to significantly impact worker or public health and safety, or have demonstrated a significant health and safety consequence.

The Haddam Neck plant was one of the earliest plant designs approved by NRC's predecessor agency, the Atomic Energy Commission. The Construction Permit was issued in May 1964 and commercial operation commenced January 1968. The facility met the construction and system design and licensing requirements then imposed by the AEC and was approved by the agency in accordance with the existing licensing process.

Since the establishment of the Atomic Energy Act of 1954, the licensing process evolved significantly over time, increasing in the level of detail considered by the staff. As described in NUREG/BR-0175[4], during the late 1950s and early 1960s the use of nuclear power to generate electricity was a novel and developing technology. In accordance with the Atomic Energy Act of 1954, the AEC did not require that a prospective power reactor owner submit finalized technical data on the safety of a facility to receive a construction permit. The agency was willing to grant a conditional permit as long as the application provided "reasonable assurance" that the projected plant could be constructed and operated at the proposed site "without undue risk to the health and safety of the public."

In this early period, at Haddam Neck (as well as other facilities), the AEC's emphasis and attention was directed toward the "front-end" of nuclear power plant safety, i.e., the safe operation of the nuclear steam supply system and the associated engineered safety features, particularly emergency core cooling systems. Accordingly, the principal inspection focus was the safe operation of the nuclear reactor and the radiological safety of plant personnel. Accordingly, the agency advanced public health and safety (and environmental protection) by assuring that reactor systems were operated and maintained properly. The "back-end" inspection activities, e.g., the examination and assessment of aspects, such as radioactive waste processing and radiological effluent monitoring and control, contributed to the overall process of determining the adequacy of plant design, operation and control, and the effectiveness of licensee performance.

Generally, AEC inspection activities were periodic audits by various agency specialists. The inspection process, relative to radiological controls, was implemented to determine if the licensee adequately established, implemented and maintained procedures to meet regulatory requirements in areas that affected radiation protection of workers, the public and the environment. Inspections were focused on assuring that fundamental aspects and specifications were met, e.g., that workers were effectively monitored for radiation exposure and the exposures were maintained within the regulatory limits, access to high radiation areas was controlled in accordance with license requirements, radiological postings and barriers were properly established and maintained, the spread of radiological contamination within the plant was controlled and monitored, appropriate surveys were conducted to support radiological work and radiological gaseous and liquid effluent releases were in conformance with regulatory requirements.

The regulatory limits specified in 10 CFR 20 were established conservatively, well below values that could affect public health and safety. To assure that these regulatory limits would not be exceeded, the licensing bases established conservative safety limits, including associated surveillance and procedural requirements. The licensee's conformance with the specified surveillance and procedural requirements was inspected regularly. Radiological waste processing and effluent control programs were generally viewed as successfully implemented if the gaseous and liquid radiological releases were maintained in accordance with the applicable regulatory requirements and technical specifications. As

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[4] "A Short History of Nuclear Regulation, 1946-1990," J. Samuel Walker, NRC Historian, Office of the Secretary, Nuclear Regulatory Commission. Published by the NRC as NUREG/BR-0175.

new General Design Criteria (GDC) and ALARA Design Objectives were later introduced by the agency in June 1974, the licensees were required to evaluate the existing systems and make modification as necessary to meet new established criteria and objectives.

In 1977, the NRC initiated a significant review of the design of older operating nuclear power plants to confirm and document their safety relative to more recent design and licensing requirements that were established in 1975[5], determine how differences should be resolved and evaluate existing plant safety. The Haddam Neck facility was selected as one of the older plants subject to the NRC's program for Integrated Plant Safety Assessment[6] in accordance with the NRC's established Systematic Evaluation Program. The assessment determined that safety margins were adequate and that the plant did not pose an undue safety risk to public health and safety. Notwithstanding, the assessment did recommend a variety of equipment modifications or additions, some changes to procedures and Technical Specifications and various engineering evaluations and design analyses.

In March 1979, NRC Region I inspection resources were diverted to accommodate agency response to the accident at Three Mile Island. Significant staff efforts were directed toward accident response and investigation, survey and radiological evaluation of the facility and surrounding environment, regulatory review, lessons-learned assessment and preparation for congressional hearings. Consequently, regulatory attention to Haddam Neck and other Region I facilities was limited during that period. In addition, while not a direct outcome of the TMI accident, the agency established its commitment to the NRC Resident Inspector Program to improve its monitoring of plant activities. The NRC resident inspector for Haddam Neck began in March 1980.

Relative to radiological control initiatives, the agency further refined the concept of As Low As Reasonably Achievable (ALARA) and developed new regulatory guidance and requirements to effect the ALARA concept in plant design, radiological effluent releases and radiological control practices. In the case of Haddam Neck, this led to enhancements in radiological environmental technical specifications and plant effluent control and monitoring practices and design. NRC inspection activities became more focused on assessing the licensee's ALARA efforts and the results achieved. Annual effluent releases decreased to typically less than one millirem -- a small fraction of the annual release limits.

More recently, inspection program implementation at Haddam Neck and other facilities has been directed toward performance-based endeavors. Inspection attention is focused toward evaluating human performance errors and the licensee's efforts to achieve remediation for recurring human performance problems; assessing material condition of equipment and facilities that have the potential to impact plant and public safety; and

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[5]Standard Review Plant, NUREG-75/087, published December 1975 and updated July 1981 as NUREG-0800.

[6]NUREG-0826, Integrated Plant Safety Assessment, Haddam Neck Plant, June 1983; NUREG-1185, Integrated Safety Assessment Report, Haddam Neck Plant, July 1987.

encouraging and promoting licensee conduct of self-assessment (i.e., problem identification, root cause determination and corrective action effectiveness).