IPRenewal NPEmails

From: John White
Sent: Tuesday, May 13, 2008 5:04 PM
To: Marsha Gamberoni; Darrell Roberts
Cc: James Noggle
Subject: FW: Indian Point: NRC Issues Indian Point Groundwater Inspection Report
Attachments: 2008-05-13 NRC IPEC Groundwater Inspection Report ML081340425.pdf; 2008-05-09 IP1 Amendment 53 Approval Ltr ML081070537.pdf; 2008-05-09 IP1 Amendment 53 SER ML081070550.pdf

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From: Ted Wingfield
Sent: Tuesday, May 13, 2008 3:58 PM
To: Beth Mizuno; Bo Pham; Brice Bickett; David Roth; Diane Screnci; Donald Jackson; Eugene Cobey; Eugene Dacus; James Trapp; John Boska; John White; Lloyd Subin; Marjorie McLaughlin; Mark Kowal; Nancy McNamara; Neil Sheehan; Rani Franovich; Richard Barkley; Richard Conte; Sammy McCarver; Ted Wingfield; Theodore Smith; Sherwin Turk; Undine Shoop; Shawn Williams
Cc: David Decker
Subject: FW: Indian Point: NRC Issues Indian Point Groundwater Inspection Report

FYI, we have implemented the following communication (key messages & documents) to external stakeholders.

VR,
Ted Wingfield
NRC, Region I
Division of Reactor Projects
Branch 2
(W) 610-337-5142
ted.wingfield@nrc.gov

From: Ted Wingfield
Sent: Tuesday, May 13, 2008 3:55 PM
To: astiebeling@pcbes.org; aws1@westchestergov.com; Catherine Borgia (borgiac@assembly.state.ny.us); chrisc@westchesterlegislators.com; ediana@co.orange.ny.us; galefs@assembly.state.ny.us ; greeleyd@co.rockland.ny.us; jefft@townofcortlandt.com; lindap@townofcortlandt.com; longon@co.rockland.ny.us; millerc@co.rockland.ny.us; robert.bondi@putnamcountyny.com; sleary@co.orange.ny.us; tlannon@pcbes.org; vob@bestweb.net
Subject: Indian Point: NRC Issues Indian Point Groundwater Inspection Report

Good afternoon,

On May 7, 2008, the NRC completed an inspection of Entergy's investigation, characterization, and actions relating to the current groundwater contamination condition at Indian Point Energy Center (IPEC) that is the result of leakage associated with the Unit 1 (U1) and Unit 2 (U2) spent fuel pool (SFP) systems.

Entergy committed to remove the spent fuel from and drain the U1 SFP by the end of 2008. Entergy is developing its long term monitoring program, which will be subject to ongoing NRC inspection.

The NRC determined that public health and safety has not been, nor is likely to be, adversely affected due to the groundwater contamination. The dose consequence to the public is a fraction of the conservatively-established NRC regulatory limits.

The NRC found Entergy’s response to the groundwater contamination to be reasonable and technically sound. The NRC identified one minor violation related to quality control of groundwater sampling.

Independent sampling and analyses conducted by the NRC and the New York State Department of Environmental Conservation confirm that contaminated groundwater is not migrating off-site except to the Hudson River. Because the affected area of the Hudson River is not a source of potable water, the principal exposure pathway to humans is from the assumed consumption of fish. No radioactivity distinguishable from background was detected during the most recent sampling and analysis of fish and crabs taken from the affected portion of the river.

RELATED ISSUE: On May 9, 2008, the NRC approved a license amendment request from Entergy to allow use of the Indian Point Unit 1 main fuel handling building crane to support the movement of spent fuel from the Unit 1 fuel pools to on-site dry storage.

Copies of the inspection report and the amendment approval letter and associated safety evaluation report are provided for your information.

VR,
Ted Wingfield
NRC, Region I
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From: John White

Created By: John.White@nrc.gov

Recipients:
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Tracking Status: None
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Recipients Received:

May 13, 2008

EA-08-088

Mr. Joseph Pollock
Site Vice President
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, NY 10511-0249

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNITS 1 & 2 - NRC INSPECTION REPORT NOS. 05000003/2007010 and 05000247/2007010

Dear Mr. Pollock:

On May 7, 2008, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Indian Point Nuclear Generating Units 1 & 2. The purpose of this inspection, initiated on November 7, 2007, was to assess your site groundwater characterization conclusions and the associated radiological significance relative to Entergy=s discovery of a small amount of contaminated water leaking from the Unit 2 spent fuel pool, and the subsequent discovery of additional subsurface groundwater contamination emanating from the Unit 1 spent fuel pool system. This inspection focused on assessing Entergy=s groundwater investigation to evaluate the extent of contamination, and the effectiveness of actions, taken or planned, to effect appropriate mitigation and remediation of the condition.

The inspection involved an examination of activities conducted under Entergy's license as they relate to safety and compliance with the Commission=s rules and regulations, and with the conditions of the license. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observations of activities, interviews with personnel, and independent analytical and assessment activities. This inspection effort reviewed Entergy's long-term monitoring plan intended for continuing verification and validation of the effectiveness of the licensee's efforts to assess, mitigate and remediate on-site groundwater conditions relative to public health and safety and protection of the environment. Details associated with the long term monitoring program will continue to be the subject of ongoing NRC inspection. The NRC will also continue split sampling for analytical comparison of selected groundwater monitoring wells through 2008. During the course of this inspection, we coordinated activities with representatives of the New York State Department of Environmental Conservation, who observed our inspection and contributed valuable expertise and independent assessment relative to its own focus on public health and safety, and environmental protection.

The enclosed inspection report documents the inspection findings, which were discussed on May 7, 2008, with Mr. Don Mayer and other members of your staff. The team found Entergy=s response to identified conditions to be reasonable and technically sound. The existence of on-site groundwater contamination, as well as the circumstances surrounding the causes of leakage and previous opportunities for identification and intervention, have been reviewed in detail. Our inspection determined that public health and safety has not been, nor is likely to be, adversely affected, and the dose consequence to the public that can be attributed to current on-site conditions associated with groundwater contamination is negligible. No significant findings were identified. However, one minor violation with respect to quality control of groundwater sampling is discussed in this report. This violation is not subject to enforcement action in accordance with Section IV of the NRC Enforcement Policy. The NRC plans no further action with regard to this matter; and no response to this letter is required.

Based on a telephone discussion between Messrs. John McCann, Director of Licensing, and Samuel Collins, NRC Region I Regional Administrator, on April 21, 2008, we understand that Entergy has committed to remove and transfer all spent fuel from the Unit 1 Spent Fuel Pool to Indian Point's Independent Spent Fuel Storage Installation, and drain the spent fuel pool by December 31, 2008, thereby essentially terminating the source of groundwater contamination from that location. Notwithstanding, it is expected that some water will remain on the bottom of the pool to reduce the potential for airborne contamination, provide shielding, and facilitate the removal of the sediment in early 2009. We understand that Entergy will promptly inform the NRC of any condition that could potentially impact or delay this commitment. Additionally, we understand that Entergy will incorporate the implementation requirements of its Long Term Monitoring Program (LTMP) as regulatory specifications in the Indian Point Energy Center's (IPEC) Off-site Dose Calculation Manual, thereby assuring that the LTMP will be regarded as an extension of the Radiological Effluents Technical Specifications and Radiological Environmental Monitoring Program, which are subject to NRC inspection. During the Exit Meeting on May 7, Entergy agreed to document these commitments to the NRC by May 20, 2008. Please inform us if our understanding is not correct.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC=s document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room). Further, in light of ongoing public interest in these matters, the NRC has scheduled a public meeting in Cortland, New York on May 20, 2008, as announced by our Meeting Notice dated May 10, 2008, also available at the NRC web site at http://www.nrc.gov/reactors.plant-specific-items/Indian-point-issues.html, to discuss NRC's assessment of Entergy's performance and actions to address the groundwater conditions at Indian Point, and the associated impact on public health and safety of the environment.

Sincerely,

/RA/

Marsha K. Gamberoni, Director
Division of Reactor Safety

Docket Nos: 50-003, 50-247
License Nos: DPR-5, DPR-26

Enclosure: Inspection Report Nos. 05000003/2007010, 05000247/2007010
w/Attachment: Supplemental Information

cc w/encl:
Senior Vice President, Entergy Nuclear Operations
Vice President, Operations, Entergy Nuclear Operations
Vice President, Oversight, Entergy Nuclear Operations
Senior Manager, Nuclear Safety and Licensing, Entergy Nuclear Operations
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P. Tonko, President and CEO, New York State Energy Research and Development Authority
C. Donaldson, Esquire, Assistant Attorney General, New York Department of Law
A. Donahue, Mayor, Village of Buchanan
J. G. Testa, Mayor, City of Peekskill
R. Albanese, Four County Coordinator
S. Lousteau, Treasury Department, Entergy Services, Inc.
Chairman, Standing Committee on Energy, NYS Assembly
Chairman, Standing Committee on Environmental Conservation, NYS Assembly
Chairman, Committee on Corporations, Authorities, and Commissions
M. Slobodien, Director, Emergency Planning
P. Eddy, NYS Department of Public Service
Assemblywoman Sandra Galef, NYS Assembly
T. Seckerson, County Clerk, Westchester County Board of Legislators
A. Spano, Westchester County Executive
R. Bondi, Putnam County Executive
C. Vanderhoef, Rockland County Executive
E. A. Diana, Orange County Executive
T. Judson, Central NY Citizens Awareness Network
M. Elie, Citizens Awareness Network
D. Lochbaum, Nuclear Safety Engineer, Union of Concerned Scientists
Public Citizen's Critical Mass Energy Project
M. Mariotte, Nuclear Information & Resources Service
F. Zalcman, Pace Law School, Energy Project
L. Puglisi, Supervisor, Town of Cortlandt
Congressman John Hall
Congresswoman Nita Lowey
Senator Hillary Rodham Clinton
Senator Charles Schumer
G. Shapiro, Senator Clinton's Staff
J. Riccio, Greenpeace
P. Musegaas, Riverkeeper, Inc.
M. Kaplowitz, Chairman of County Environment & Health Committee
A. Reynolds, Environmental Advocates
D. Katz, Executive Director, Citizens Awareness Network
S. Tanzer, The Nuclear Control Institute
K. Coplan, Pace Environmental Litigation Clinic
M. Jacobs, IPSEC
W. Little, Associate Attorney, NYSDEC
M. J. Greene, Clearwater, Inc.
R. Christman, Manager Training and Development
J. Spath, New York State Energy Research, SLO Designee
A. J. Kremer, New York Affordable Reliable Electricity Alliance (NY AREA)

Docket Nos: 50-003, 50-247
License Nos: DPR-5, DPR-26

Enclosure: Inspection Report Nos. 05000003/2007010, 05000247/2007010
w/Attachment: Supplemental Information

U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos.	50-003, 50-247
License Nos.	DPR-3, DPR-26
Report Nos.	05000003/2007010 and 05000247/2007010
Licensee:	Entergy Nuclear Northeast
Facility:	Indian Point Nuclear Generating Station Units 1 & 2
Location:	295 Broadway Buchanan, NY 10511-0308
Dates:	November 7, 2007 - May 7, 2008
Inspectors:	J. Noggle, Sr. Health Physicist, CHP, team leader T. Nicholson, Sr. Technical Advisor for Radionuclide Transport J. Williams, U.S. Geological Survey, Troy, New York J. Kottan, State Agreements Officer J. Commiskey, Health Physicist
Approved by:	John R. White, Chief Plant Support Branch 2 Division of Reactor Safety

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SUMMARY OF FINDINGS

IR 05000247/2007010 & IR 05000003/2007010; 11/08/2007 - 05/07/2008; Indian Point Nuclear Generating Station Units 1 & 2; Other Activities – associated with ROP deviation memorandum.

The report covers an inspection of a September 1, 2005, licensee-identified Unit 2 spent fuel pool leak investigation final report and long term monitoring plan; and review of historical leakage involving the Unit 1 spent fuel pool by three regional inspectors, one headquarters hydrology specialist, and a U.S. Geological Survey hydrology specialist. The NRC=s program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, AReactor Oversight Process," Revision 4, dated December 2006.

A. <u>NRC - Identified and Self-Revealing Findings</u>

No findings of significance were identified.

B. <u>Licensee - Identified Violations</u>

None

EXECUTIVE SUMMARY

Background:

On September 1, 2005, the NRC was informed by Entergy that cracks in a Unit 2 spent fuel pool wall had been discovered during excavation work, and that low levels of radioactive contamination were found in water leaking from the cracks having radionuclides similar to Unit 2 spent fuel pool water. Entergy initiated a prompt investigation to determine the extent of the condition and potential impact on health and safety. Initially, Entergy determined that on-site groundwater in the vicinity of the Unit 2 facility was contaminated with tritium as high as 200,000 picocuries per liter of water (about ten times the EPA drinking water standard). Subsequently, Entergy initiated actions to perform a comprehensive groundwater site characterization to investigate the extent of on-site groundwater contamination, identify the sources, and mitigate and remediate the condition. This effort required the establishment of several on-site groundwater monitoring wells to characterize groundwater behavior, flow, direction, and migration pathways.

On September 20, 2005, Region I initiated a special inspection of this matter to examine the licensee's performance and determine if the contaminated groundwater effected, or could effect, public health and safety. On October 31, 2005, NRC's Executive Director of Operations (EDO) authorized continuing NRC inspection to assess licensee performance of on-site groundwater investigation activities, and independently evaluate and analyze data and samples to assure the effectiveness and adequacy of the licensee's efforts. Throughout this effort, the NRC coordinated its inspection activities with the New York State Department of Environmental Conservation (DEC), which initiated its own independent assessment of the groundwater conditions, including observation of NRC's inspection activities.

The NRC issued a special inspection report on March 16, 2006 (ADAMS Accession No. ML060750842). The report assessed Entergy's performance, achievements, and plans relative to radiological and hydrological site characterization; and reported that the on-site groundwater contamination did not, nor was likely to, adversely affect public health and safety. In the report and in subsequent public meetings, NRC indicated that it would continue to inspect licensee performance in this area, including independent evaluation and analysis of data, to assure that Entergy continued to conform to regulatory requirements, and that public health and safety was maintained.

On March 21, 2006, NRC's independent on-site groundwater sample analysis effort first determined that strontium-90 was also a contaminant in the groundwater, a fact that was subsequently confirmed by Entergy and the DEC. This determination resulted in a significant expansion of the on-site groundwater characterization effort since the source of the strontium-90 contaminant was traced to leakage from the Unit 1 Spent Fuel Pool. A full site-wide hydrogeologic investigation was subsequently scoped to include Unit 1 and Unit 3. The NRC inspection charter objectives were similarly revised to provide the necessary oversight. Off-site groundwater samples have also been obtained since the fall of 2005, and have never detected any off-site groundwater contamination.

Since that time, the NRC has continued to inspect and monitor Entergy's activities beyond the limits of normal baseline inspection, as authorized by NRC's Executive Director of Operations (EDO). During this period, NRC inspectors closely monitored Entergy's groundwater characterization efforts, and performed independent inspection of radiological and hydrological conditions affecting on-site groundwater. Additionally, from early 2006 through January 2008, the NRC kept interested Federal, State, and Local government stakeholders informed of current conditions through routine bi-weekly teleconferences.

Status of Current Activities, Plans, and Inspection Results:

On January 11, 2008, Entergy submitted the results of its comprehensive ground water investigation, and included its plan for remediation and long-term monitoring of the on-site groundwater conditions. In its report, Entergy described the sources of the groundwater contamination to be the Unit 1 and Unit 2 spent fuel pools. While both pools contributed to the tritium contamination of groundwater, leakage from the Unit 1 spent fuel pool was determined to be the source of other contaminants such as strontium-90, cesium-137, and nickel-63. Entergy identified its plan to remove all fuel from the Unit 1 spent fuel pool to an on-site storage location and drain the spent fuel pool system by the end of 2008, thereby essentially eliminating the source of the groundwater contamination from that facility. Some water is expected to remain in the bottom of the pool to reduce the potential for airborne contamination and provide shielding until the residual sludge is removed in early 2009. In the January 11, 2008 report, Entergy described its actions to repair or mitigate all identified potential leak locations in the Unit 2 spent fuel pool system that may have contributed to the on-site tritium-contaminated groundwater in the vicinity of that facility.

Notwithstanding, residual radioactivity is expected to continue to impact on-site groundwater for the duration of licensed activities. On-site groundwater is expected to continue to be monitored and reported as an abnormal liquid release in accordance with NRC regulatory requirements. No off-site groundwater has been impacted, since the on-site groundwater flow is to the discharge canal and the Hudson River. Accordingly, the licensee has established a long-term monitoring strategy for the purpose of evaluating the effect and progress of the natural attenuation of residual contamination, informing and confirming groundwater behavior as currently indicated by the existing site conceptual model, and determining changes in conditions that may be indicative of new or additional leakage.

Entergy's performance and effectiveness relative to successfully draining water from the Unit 1 spent fuel pool system by the end of 2008, and the quality and effectiveness of its long-term monitoring program, will be the immediate focus of NRC's continuing inspection of Entergy's performance and conformance with regulatory requirements relative to the existing groundwater conditions. Additionally, NRC will continue to inspect the efficacy of the licensee's long-term monitoring program as part of the Reactor Oversight Process pertaining to radiological environmental and effluents inspection activities.

Notwithstanding, radiological significance from the groundwater conditions at Indian Point is currently, and is expected to remain negligible with respect to impact on public health and safety and the environment. NRC has confirmed with the New York State Department of Health, that drinking water is not derived from groundwater or the Hudson River in the areas surrounding or

influenced by effluent release from Indian Point. Accordingly, the only human exposure pathway
of merit is from the possible consumption of aquatic foods from the Hudson River, such as fish and invertebrates. Dose assessment of the potential for exposure from this pathway, continues to indicate that the hypothetical maximally exposed individual would be subject to no more than a very small fraction of the NRC regulatory limit for liquid radiological effluent release.

Status of Current Inspection Results:

1. Upon the initial identification of conditions that provided evidence of an abnormal radiological effluent release affecting ground water, the licensee implemented actions that conformed to the radiological survey requirements of 10 CFR 20.1501 to ensure compliance with dose limits for individual members of the public as specified in 10 CFR 20.1302, including: (1) promptly investigating and evaluating the radiological conditions and potential hazards affecting groundwater conditions, on- and off-site; (2) annually reporting the condition, and determining that the calculated hypothetical dose to the maximally exposed member of the public was well below established NRC regulatory requirements for liquid radiological release; (3) confirming, through off-site environmental sampling and analyses, that plant-related radioactivity was not distinguishable from background; (4) initiating appropriate actions to mitigate and remediate the conditions to assure that NRC regulatory dose limits to members of the public and the environment were not exceeded; and (5) developing the bases for a long-term monitoring program to ensure continuing assessment of groundwater effluent release and reporting of the residual radioactivity affecting the groundwater. Additional refinement of the long term monitoring program is expected to occur as data is collected and evaluated to verify and validate the effectiveness of expected natural attenuation of the existing groundwater plumes, and to ensure the timely detection of new or additional leakage affecting ground water.

2. The determination of contaminated on-site groundwater conditions at Indian Point was the result of the licensee’s investigation of potential leakage from the Unit 2 Spent Fuel Pool initiated in September 2005, and subsequent development and application of a series of ground water monitoring wells to determine the extent of that condition. No evidence was found that indicated that the events at Indian Point, that resulted in the on-site groundwater contamination (identified to the NRC on September 1, 2005), were the result of the licensee’s failure to meet a regulatory requirement or standard, where the cause of the condition was reasonably within the licensee’s ability to foresee and correct, and should have been prevented. This determination is based on: interviews with licensee personnel; comprehensive review of pertinent documentation, including previous condition reports, survey records, radiological liquid effluent and environmental monitoring reports, records of historical spills and leaks documented in accordance with 10 CFR 50.75, “Reporting and Recordkeeping for Decommissioning Planning”; and extensive on-site NRC inspection to confirm licensee conformance with required regulatory requirements.

3. The current contaminated groundwater conditions at Indian Point Energy Center are the result of leakage associated with the Unit 1 and Unit 2 spent fuel pool (SFP) systems. No other systems, structures, or components were identified as contributors to the continuing on-site contamination of ground water.

4. Entergy's hydrogeologic site characterization studies provided sufficiently detailed field observations, monitoring, and test data which supported the development and confirmation of a reasonable conceptual site model of groundwater flow and transport behavior. An independent analysis of groundwater transport through fractured bedrock utilizing geophysical well logging data was conducted by the U.S. Geological Survey (USGS). The USGS assessment corroborated the groundwater transport characteristics that were determined by Entergy's contractor.

5. Entergy's hydrogeologic site characterization and developed conceptual site model provide a reasonable basis to support the determination that the liquid effluent releases from the affected spent fuel pool systems migrate in the subsurface to the west, and partially discharge to the site's discharge canal, with the remainder moving to the Hudson River. Current data and information indicates that contaminated groundwater from the site does not migrate off-site except to the Hudson River. This conceptual site model of groundwater behavior and flow characteristics is supported by the results of independent groundwater sampling and analyses conducted by NRC, which have not detected any radioactivity distinguishable from background in the established on-site boundary monitoring well locations, or in various off-site environmental monitoring locations.

6. Currently, there is no drinking water exposure pathway to humans that is affected by the contaminated groundwater conditions at Indian Point Energy Center. Potable water sources in the area of concern are not presently derived from groundwater sources or the Hudson River, a fact confirmed by the New York State Department of Health. The principal exposure pathway to humans is from the assumed consumption of aquatic foods (i.e., fish or invertebrates) taken from the Hudson River in the vicinity of Indian Point that has the potential to be affected by radiological effluent releases. Notwithstanding, no radioactivity distinguishable from background was detected during the most recent sampling and analysis of fish and crabs taken from the affected portion of the Hudson River and designated control locations.

7. The annual calculated exposure to the maximum exposed hypothetical individual, based on application of Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Release of Reactor Effluents for the Purpose of Evaluation Compliance with 10 CFR Part 50, Appendix I," relative to the liquid effluent aquatic food exposure pathway is currently, and expected to remain, less than 0.1 % of the NRC's "As Low As is Reasonably Achievable (ALARA)" guidelines of Appendix I of Part 50 (3 mrem/yr total body and 10 mrem/yr maximum organ), which is considered to be negligible with respect to public health and safety, and the environment.

8. All identified liner flaws in the Unit 2 spent fuel pool, and the initially identified crack affecting the Unit 2 spent fuel pool system have been repaired or mitigated. However, not all Unit 2 fuel pool surfaces are accessible for examination. No measurable leakage is discernable from evaporative losses based on Unit 2 fuel pool water makeup inventory data. Unit 1 spent fuel pool water is being processed continuously to reduce the radioactive concentration at the source prior to leakage into the groundwater, and actions have been initiated to effect the complete removal of spent fuel and essentially all the water from the Unit 1 Spent Fuel Pool system by the end of 2008, thereby terminating the source of 99.9% of the dose significant strontium-90 and nickel-63

contaminants (the remaining 0.1% is represented by the Unit 2 and Unit 1 hydrogen-3 (tritium) contaminants). Entergy's selected remediation approach for the contaminated groundwater conditions appears reasonable and commensurate with the present radiological risk.

9. The historical duration of leakage from the Unit 1 and Unit 2 spent fuel pool systems that resulted in groundwater contamination is indeterminate. The evidence indicates that the volume of leakage was small compared to the available water inventory, and was much less than the normally expected evaporative losses from spent fuel pools. This conclusion is based on NRC staff review and assessment of spent fuel pool makeup inventory records and applicable leakage collection data, the results of the continuously implemented Radiological Environmental Monitoring Program affecting the Indian Point site, and evaluation of the developed hydrogeologic groundwater transport model. Accordingly, there is no evidence of any significant leak or loss of radioactive water inventory from the site that was discernable in the off-site environment.

10. No releases were observed or detected from Unit 3.

11. The conditions surrounding the leaking Unit 1 spent fuel pool are based on a leakage rate of 10 drops per second (about 25 gallons per day) that was identified in 1992. At that time, the licensee performed a hypothetical bounding dose impact that concluded that there was negligible dose impact to the public caused by this condition. This licensee assessment was inspected and evaluated, at that time, by NRC inspectors. This early bounding hypothetical calculation agrees with the dose impact now confirmed by the recently completed hydrogeologic site investigation, and NRC's independent assessment. Based on extensive review of the circumstances and inspection records from that period, it appears that the licensee was in conformance with the standards, policy, and regulatory requirements that prevailed at that time.

REPORT DETAILS

4.0 OTHER ACTIVITIES (OA)

4OA5 Other Activities

.1 <u>Overview of the Groundwater Contamination Investigation</u>

In September 2005, a crack was discovered leaking on the outside of the Unit 2 spent fuel pool south wall (approximately 30 feet below the top) during excavation of the spent fuel building loading bay. The NRC initiated a special inspection on September 21, 2005, to investigate the implications of the observed Unit 2 spent fuel pool leakage. Based on analysis of the radionuclide concentrations in the Unit 2 spent fuel pool and maximum bounding pool makeup losses, a bounding dose calculation based on direct release to the Hudson River indicated a tiny fraction of 1 mrem (0.00002 mrem/yr) as the estimated dose to the maximally exposed hypothetical individual. Though the radiological significance of the circumstance was negligible, the condition was unexpected. Accordingly, NRC Region I was authorized by the Executive Director of Operations (EDO) to conduct additional oversight inspection of licensee performance and the circumstances surrounding this contamination issue to better understand the condition and examine possible generic implications, since similar conditions had been identified at other facilities.

Due to the complicated nature of the groundwater characterization effort at Indian Point (i.e., a relatively small site containing two operating units and one unit in SAFSTOR, built on a complex fractured bedrock foundation that required sophisticated analysis and modeling to fully understand groundwater behavior), the EDO renewed the increased inspection authorization each year to permit active and frequent inspection oversight. As a result, inspection of the Indian Point contaminated groundwater conditions evolved to include not only radiological environmental and effluent expertise from Region I, but also hydrological assessment expertise from NRC's Office of Research, and later, from the US Geological Survey (USGS). The application of such resources permitted the NRC to conduct several independent reviews and assessments of data, information, and analysis on which the licensee based its conclusions and determinations.

In addition, the NRC and USGS specialists, worked closely with the New York State Department of Environmental Conservations (NYS DEC) by sharing data and assessment information, coordinating independent split sampling of various sample media, and providing a combined oversight of licensee performance.

On November 7, 2005, the licensee began installing a series of monitoring wells on-site, based on an initial understanding of on-site groundwater flow patterns and associated contaminant transport. Thirty-six monitoring wells were installed over the next 2 years, with the final well installed and operational by the end of August 2007. The groundwater monitoring network ultimately developed by Entergy includes these plus a number of previously existing monitoring locations. Various geophysical evaluations and analyses, including groundwater table mapping, ground permeability measurements and groundwater gradient calculations, were performed and two site-wide hydrology tests were conducted to observe groundwater response in a network of monitoring wells. These tests included a 3-day duration groundwater pump-down test from the Unit 2 spent fuel pool (SFP) leak location, and injection of a tracer dye at the base of the Unit 2 SFP to trace its path across the site.

This body of information was utilized by Entergy to determine the sources of the groundwater contamination, evaluate the potential for leak mitigation through pumping, and confirm the site groundwater transport model through a final tracer test. Throughout the investigation frequent iterations were made to refine the extent of groundwater contamination, the total amount of contaminant released to the environment, and the resulting public dose assessment to ensure that public health and safety were maintained.

As additional wells were drilled and sampled, gradually the full extent of on-site ground water contamination was revealed. A short synopsis providing the significant highlights of the licensee's investigation follows, with a more detailed timeline provided in Attachment 1, "Timeline Synopsis".

On February 27, 2006, hydrogen-3 (tritium) contamination was detected in a monitoring well beyond the discharge canal, providing the first evidence of potentially contaminated groundwater being directly released into the Hudson River. On February 28, 2006, the licensee developed a new groundwater release bounding calculation methodology based on an overall site rainfall recharge into several discrete site drainage areas to the Hudson River. On March 21, 2006, radionuclides other than tritium (strontium-90 and nickel-63) were first discovered in a monitoring well, which was later determined to be associated with the Unit 1 spent fuel pool system.

On April 24, 2006, utilizing a rainfall recharge water mass balance approach to calculate groundwater flow and more recent monitoring well data utilizing the maximum concentrations of hydrogen-3 (tritium), strontium-90, and nickel-63, a new revised public dose estimate (from the hypothetical consumption of fish) indicated a maximum hypothetical public dose of 0.0025 mrem/yr to the total body and a maximum of 0.011 mrem/yr to the highest organ (adult bone). These values represent about 0.1% of the regulatory specification for liquid effluent releases contained in the Offsite Dose Calculation Manual. This specification is derived from 10CFR50, Appendix I, As Low As is Reasonably Achievable (ALARA) design objectives for liquid effluent releases.

The basis for calculating public doses is site specific, and at Indian Point, is based on the hypothetical, assumed consumption of fresh water fish and salt water invertebrates. Due to a higher dose significance of strontium-90 detected in groundwater releases, Entergy revised its Off-site Dose Calculation Manual (ODCM) to include the analysis of strontium-90 in environmental media, such as fish and invertebrates collected from the Hudson River. Consumption of fish was assumed notwithstanding the fact that the New York State Department of Health publishes health advisories for sport and game fish and recommends very limited or no consumption of fish be taken from the lower reaches of the Hudson River due to mercury and Poly-Chlorinated Biphenyls (PCB) contaminants.

Subsequently, during the summer of 2006, Entergy collected and analyzed fish from the Hudson River, and strontium-90 was identified in one fish collected near the plant as well as in several fish caught in a control location 20 miles upstream of the plant at similar concentrations. In order to resolve whether the strontium-90 was plant-related or the result of existing background levels (Sr-90 exists in environment due to weapons-related fallout), an expanded fish sampling program was devised by the New York State DEC. The program included an additional 90 mile upstream sample location, the collection of specific fish species identified by the State's biologist as having limited migratory behavior, and a three-way split of the edible fish portions of the prepared samples between NRC, Entergy, and the NYS DEC. The effort was conducted in June 2007. In the expanded samples, all three independent analytical laboratories reported results that indicated that no plant-related radioactivity was detected or distinguishable from background. To date, no offsite environmental samples (other than water samples from the discharge canal and the tidally influenced intake structure) have indicated any detectable plant-related radionuclides,

The USGS performed an independent fracture flow analysis to determine on-site groundwater flow utilizing different data and methods than Entergy to compare groundwater flow results with the licensee. This provided a comparison of fracture flow dominated groundwater flow with the licensee's groundwater flow results based on an assumption of general porous media flow through dense fracture sets in the ground. No significant differences were observed from these comparisons, which essentially confirmed that either model of groundwater transport flow provided valid results.

On January 11, 2008, Entergy submitted a hydrogeologic site investigation final report to the NRC documenting closure of the groundwater investigation, adoption of selected remediation actions, and a plan for the continued long-term monitoring of the existing contaminant plumes (ADAMS Accession No. ML080320600). On January 25, 2008, Entergy submitted a synopsis of the long term monitoring plan basis to describe a groundwater monitoring network and a sampling schedule to continue monitoring the existing plumes, detect any future Unit 2 spent fuel pool leaks, and detect any future leaks from any other plant systems structures or components at the site (ADAMS Accession No. ML080290204).

This inspection report provides NRC review of the above mentioned licensee activities. Continued NRC inspection will continue through 2008 of the removal of spent fuel and draining of the leaking Unit 1 spent fuel pool, split sampling to verify the basis of licensee's off-site dose assessment, and review of further development and refinements to the licensee's long term monitoring plan. Inspection findings will be documented in future reports.

.2 <u>Final Groundwater Contamination Characterization</u>

By the end of 2007, based on over 900 monitoring well samples, the extent of the on-site subsurface contamination had been mapped and the sources have been determined. Two on-site plumes were discovered emanating from the Unit 2 and Unit 1 spent fuel pool regions, respectively. Due to the influence of the Unit 1 building foundation drain system, some of the Unit 2 plume was drawn into the Unit 1 area, with both plumes intermingling

and following a converging path westward towards the Hudson River. Both plumes were relatively shallow (less than 200 feet below ground surface) following a common groundwater trough between Units 1 and 2, and a groundwater transport velocity of between 4 and 9 feet per day, covering a total distance of about 400 feet to the Hudson River (see Figure 1). Approximately one-half of the combined plumes are being intercepted by the plant discharge canal which allows for substantial dilution of this fraction and is a monitored discharge path. The other portion of the combined plumes flows below the discharge canal and discharges directly into the bottom of the Hudson River.

Due to limited groundwater sampling of the new river front monitoring wells across normal seasonal groundwater flow variations, no trend in plume concentrations is yet discernable. Current contaminant concentrations detected from monitoring wells closest to the Hudson River indicate 9,000 pCi/L of hydrogen-3 (tritium) and 27 pCi/L of strontium-90. A map of monitoring well locations and a table of radionuclide concentration values at each monitoring well are provided in Attachment 2.

These concentrations are slightly below the minimum required effluent release detection sensitivities for these radionuclides (i.e., 10,000 pCi/L for hydrogen-3 (tritium) and 50 pCi/L for strontium-90), and well below the maximum allowable liquid effluent release ALARA guidelines of ten times the effluent concentrations in 10 CFR 20, Appendix B, Table 2, Column 2 (10,000,000 pCi/L for hydrogen-3 (tritium) and 5,000 pCi/L for strontium-90). NRC required calculation of the maximum dose to a hypothetical person consuming fish and invertebrates at the site boundary, indicates less than 0.1% of design objectives for liquid effluents (3 mrem total body and 10 mrem maximum organ). Since the groundwater contamination is considered an abnormal release, the condition is required to be quantified, evaluated and reported in the annual radiological effluent release reports.

.3 Groundwater Sampling

a. Inspection Scope

During the licensee's groundwater investigation, over 900 groundwater samples were collected and analyzed from the established on-site monitoring well network by the end of 2007. The analytical results provide the basis for assessing the extent of the groundwater plume and for performing calculations of offsite doses to members of the public. In order to assess Entergy's performance in this area, the NRC implemented an independent split sample collection program with the licensee beginning in September 2005. The monitoring wells selected for independent verification included the southern boundary wells and those bordering the Hudson River that were utilized in effluent release and dose assessment calculations. Sample identity was assured by chain-of-custody procedures that included sample collection observation by the NRC or a representative of the NYS DEC. The NRC samples were analyzed by an independent government laboratory. The NRC samples were sent to the NRC contract laboratory, the Oak Ridge Institute for Science and Education (ORISE), Environmental Site Survey and Assessment Program

(ESSAP) radioanalytical laboratory.

By the end of 2007, over 250 split groundwater samples were obtained to provide an independent check of Entergy's analytical results and to independently verify if there was any detectable migration of groundwater contaminants offsite. These split samples represent over 1,000 analyses, primarily for hydrogen-3 (tritium), strontium-90, nickel-63, and gamma-emitting radionuclides that characterized the effluent releases. Analyses for other radionuclides were performed, but none were detected.

Various in-plant contamination sources (the Unit 1 and 2 spent fuel pools and others) were also sampled and analyzed by the NRC for a complete range of radionuclides to evaluate the known and potential leaking sources of radioactivity, and to ensure an adequate scope of radionuclide analysis was conducted by the licensee in their groundwater sampling campaign. In addition, the NRC analyzed miscellaneous environmental samples of interest including offsite water supply sources, Hudson River aquatic vegetation, and fish samples. The New York State DEC also provided confirmation of the licensee's sample analysis results through a parallel split sample program. This provided for a three-way laboratory comparison of many of the offsite release and environment-critical sample results. This three-way data comparison provided for timely identification of any discrepant sample results potentially affecting offsite releases.

b. <u>Findings and Assessment</u>

No findings of significance were identified.

In general, Entergy=s groundwater measurements of radioactivity were of good quality and of sufficient sensitivity to assess radiological impact. The quality of Entergy=s measurements were confirmed by various split samples analyzed by NRC and the State of New York, (i.e., the Department of Environmental Conservation and the Department of Health). Of the over 1000 results that were reviewed, there were some sample disagreements based on the statistical comparison criteria specified in NRC Inspection Procedure 84750, "Radioactive Waste Treatment, and Effluent and Environmental Monitoring." A discussion of the sample disagreements is provided below.

- Between March and September 18, 2006, Entergy reported some strontium-90 results associated with the Unit 1 plume that were low when compared to NRC results. Entergy's results indicated that the Unit 1 spent fuel pool cleanup system had shown a reduction in the associated groundwater plume concentrations over a relatively short period of time. There was no other consequence due to this disparity. Entergy initiated an investigation into this issue with their offsite contract laboratory. The investigation did not identify a definitive cause. As a result, Entergy terminated its contract with the lab and procured the services of another offsite laboratory. Entergy's reanalysis of the samples confirmed that the original results were low. The reanalysis results were subsequently in agreement with the NRC laboratory results.

- Entergy reported no detectable nickel-63 contamination in four samples from Monitoring Well-42 taken on November 16-17, 2006. Since Monitoring Well-42 is closest to the Unit 1 SFP, and other radionuclides analyzed at the same location remained at expected levels, this indication was not considered reasonable and was also not in agreement with the New York State or NRC laboratory results. This resulted in an investigation into this issue by the licensee's new off-site contract laboratory. Improper procedure protocol was identified and additional controls were implemented to correct this issue. Reanalysis of the nickel-63 results were in agreement with the NRC laboratory results. No other significant sample anomalies were identified by the NRC through the end of 2007.

The above NRC-identified discrepancies highlighted the need for quality control in the licensee's sample acquisition and laboratory processing and measurement processes. Oversight of offsite laboratory analysis of samples was not originally specified by the licensee for on-site groundwater sampling. NRC radiological environmental monitoring program laboratory quality control requirements, specify radionuclide detection sensitivities, and require blind blank samples and blind radionuclide-spiked samples to be provided by the licensee as a check on the off-site laboratory's analytical performance. These requirements apply to the offsite radiological environmental monitoring program, but no requirements are specified for on-site groundwater sample quality controls.

NRC radiological effluent sampling analyses also require laboratory quality controls as specified above. On February 27, 2006, based on detecting hydrogen-3 (tritium) in a monitoring well near the Hudson River, Entergy revised their bounding dose calculation and began calculating actual effluent releases via the groundwater pathway. At this point in the groundwater investigation, the quality assurance of groundwater sample analyses used in effluent reporting became a requirement. However, the offsite laboratory analyses of groundwater samples were not independently evaluated by Entergy until more than one year later. Technical Specifications Section 5.4.1(a) specifies written procedures shall be established, implemented, and maintained covering Appendix A of Regulatory Guide 1.33, Revision 2, which specifies quality assurance requirements for procedures associated with the control of radioactive effluents released to the environment. The inadequate procedure (O-CY-1420, Rev. 1), constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC Enforcement Policy. There was no actual or potential consequence of this procedure deficiency, because in function, the NRC and NYS DEC split sampling program provided a very effective verification of Entergy's laboratory sample analysis program during the groundwater investigation by assuring the accuracy of analytical results.

To address this concern, in May 2007, Entergy initiated an on-site groundwater sampling quality control program incorporating a blind blank sample and blind radionuclide-spiked sample program to verify its own offsite laboratory analytical results. In addition, Entergy's corrective action program is still addressing the quality control program requirements relative to groundwater sample analysis, with corrective action responsibilities transferred to the corporate group for resolution (CR-HQN-2007-00894).

NRC split sample analysis comparison of the licensee's groundwater sample results are expected to continue until such time as Entergy has addressed all of the concerns associated with laboratory quality assurance issue.

Due to the presence of strontium-90 in groundwater monitoring wells close to the Hudson River, Entergy modified their environmental monitoring analysis of fish samples to include strontium-90 analysis and in September 2006, strontium-90 was detected in one of six fish caught near the plant. Three out of six samples caught 20 miles upstream at the control location also contained similar detectable levels of strontium-90. Entergy concluded that no strontium-90 was detected above background based on similar results obtained from the control location. Strontium-90 is not uniquely generated by nuclear power plants, but was also generated from above ground nuclear testing in the early 1950's and 1960's and now exists ubiquitously in the environment. From a review of applicable scientific literature, comparable levels of strontium-90 that were detected in the September 2006 fish samples were also indicated in background fish testing results in other parts of New York State.

To further clarify the origin of the strontium-90 and confirm the efficacy of utilizing Entergy's control location in monitoring background strontium-90 concentrations in fish, an expanded fish sampling program was conducted in June 2007 led by NYS DEC, in consultation with its fish biologists, to ensure that the control location is sufficiently removed from Indian Point to preclude fish migration and to accurately represent background levels of strontium-90. This expanded fish sampling program collected fish samples from three Hudson River locations: an area influenced by liquid releases from Indian Point, a control location 20 miles upstream, and a special control location 90 miles upstream in the Catskills. Three-way split fish samples were supplied to Entergy, NYS DEC and NRC for inter-laboratory comparison of these results. Neither strontium-90 nor any plant-related radionuclides were detected in any edible fish samples by any of the three participating laboratories at any of the three Hudson River locations. This is considered significant, since public doses from liquid discharges from Indian Point are calculated based on assumed fish and invertebrate consumption. This confirms the results expected from the groundwater effluent and normal plant liquid effluent release calculations, indicating small fractions of one millirem per year to the maximally exposed hypothetical member of the public that consumes fish and invertebrates.

.4 Dose Assessment

a. Inspection Scope

Groundwater effluent discharges and associated hypothetical dose calculations to the public involve a two-step process. First, a groundwater transport model is developed to estimate the amount of radioactive material being discharged and its dilution into the environment. The hydrogeologic site investigation of Indian Point has provided the results for determining this aspect of the dose calculation.

Second, based on methods defined in the Indian Point Energy Center Offsite Dose Calculation Manual (ODCM), calculations are performed to determine the maximally exposed individual (infant, child, teen or adult) and maximum organ (bone, kidney, gastro-intestinal tract, liver, thyroid, lung and total body). NRC has confirmed with the NYS Department of Health that groundwater and Hudson River water is not used for drinking or irrigation purposes in the area surrounding Indian Point Energy Center. Therefore, at Indian Point Energy Center, the liquid effluent dose pathway is through the ingestion of fish and invertebrates (crab). Both the groundwater effluent discharge and the pathway-to-man methodologies and calculation methods were reviewed throughout the licensee’s investigation in order to ensure that the significance of the liquid effluent releases were bounded and the associated dose impact was evaluated to provide an accurate dose assessment of public health and safety.

b. Findings and Assessment

No findings of significance were identified.

The licensee performed an initial conservative bounding dose calculation, dated October 21, 2005, that assumed a worst case condition, i.e., Unit 2 spent fuel pool water being discharged directly into the Hudson River with minimal Hudson River dilution flow (approximately 100,000 gallons per minute). This dose assessment assumed a conservative Unit 2 SFP leak rate of 2.6 gallons per day[1] incorporating all the radionuclides detected. The resultant calculated dose was about 0.0001 millirem/year, well below the ALARA design objectives for liquid effluent releases (3 millirem/year per reactor) and a very small percentage of the public dose limits (100 millirem per year).

The inspectors concluded that the licensee's preliminary offsite dose calculation utilized conservative assumptions regarding the Unit 2 SFP leak rate and groundwater dilution, appropriately applied the methodology of the licensee's Offsite Dose Calculation Manual, provided a timely dose evaluation response to the identified condition.

As more data became available, the licensee performed a revision to the conservative bounding calculation, dated December 13, 2005, using Hudson River dilution based on a six hour half-tidal surge. This resulted in a dilution volume of 1.45E10 gallons. This revised bounding dose calculation was based on the actual radioactivity concentration of the Unit-2 SFP and the resultant annual dose to the hypothetical maximally exposed member of the public was calculated to be about 0.0001millirem/year. This revision was based on conservative and reasonable assumptions and agreed with the result from the original bounding calculation.

As on-site groundwater monitoring wells were installed, groundwater sample results were collected, water table contours were identified, and groundwater transport parameters were determined. Entergy developed a site area drainage model based on annual rainfall groundwater recharge water balance and applied maximum monitoring well groundwater concentrations, which was used in a February 28, 2006 effluent release and off-site dose calculation with a result of 0.000015 mrem/yr to the maximally

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[1]The basis for the assumed value of 2.6 gallons per day is discussed in Section 5 of this report.

exposed hypothetical member of the public. This was no longer a bounding calculation, but
represented an actual groundwater effluent release determination based on groundwater measurements and groundwater drainage calculations. Radiological and hydrogeologic inspection of this method determined that the basis was reasonable and the calculations were accurate.
Later in the investigation on March 21, 2006, NRC sample results of Monitoring Well-37 (a river front monitoring well) indicated strontium-90 concentration of 26 pCi/L. This was the first indication that strontium-90 was likely being released directly to the Hudson River through the groundwater. Licensee results confirmed both strontium-90 and nickel-63, in addition to hydrogen-3 (tritium), were likely migrating to the Hudson River. The dose significance for these additional radionuclides is over one hundred times that of hydrogen-3 (tritium). On April 24, 2006, Entergy updated their dose assessment in recognition of this new monitoring well data, and applied the maximum concentrations of hydrogen-3 (tritium), strontium-90 and nickel-63. The resulting groundwater effluent discharge and off-site dose assessment indicated a maximum hypothetical public dose of 0.0025 mrem total body and 0.011 mrem maximum organ dose (adult bone) per year. The increase from the previous dose estimates is a direct result of the strontium-90 and nickel-63 radionuclides.

As additional groundwater sample data became available, the licensee's dose assessment model was further refined to rank the monitoring well sample data in each site drainage area from low to high, and apply a 75th percentile of radionuclide concentration to the dose assessment calculations. This approach was determined to be more realistic and yet still conservative. Utilizing this methodology, abnormal groundwater effluent releases were calculated and the following doses for groundwater releases in 2005 and 2006 were officially reported to the NRC in the annual radiological effluent release reports as follows:

2005: 0.00212 mrem total body and 0.0097 mrem maximum organ (adult bone)
2006: 0.00178 mrem total body and 0.0072 mrem maximum organ (adult bone)

Based on discussions with the NRC and USGS hydrologists, Entergy agreed to further evaluate the groundwater flow rate model to utilize groundwater flux calculations based on Darcy's Law, a hydrogeological algorithm that considers actual groundwater gradient and soil permeability rather than inferring groundwater flow based on a rainfall infiltration model. Accordingly, Entergy initiated actions to develop a refined method to calculate local drainage area groundwater flux calculations based on Darcy's Law while retaining an overall rainfall infiltration as input to the local drainage calculations. Entergy intends to use this approach to calculate and report the 2007 groundwater effluent discharges and dose assessments.

.5A <u>Unit 2 SFP Leakage</u>

a. <u>Inspection Scope</u>

The Unit 2 SFP does not have a leak detection system, therefore, the licensee used alternative means of assessing the amount of leakage from the spent fuel pool.

Detectable fuel pool inventory loss could not be determined based on fuel pool water makeup records, given the variability in water evaporation loss due to atmospheric temperature, pressure, and humidity variations. A more sensitive indicator of spent fuel pool water loss utilized the trending of spent fuel pool boric acid concentration over time, since boric acid is not affected by evaporative losses and any reduction in boric acid concentration would likely be due to leakage.

The NRC followed Entergy's progress in examination of the Unit 2 SFP liner and transfer canal for leaks and subsequent repair of a through-wall leak in the transfer canal.

As was reported in the March 16, 2006 special inspection report, NRC investigation into the capture efficiency of the Unit 1 building foundation drain system indicated approximately seven times more hydrogen-3 (tritium) radioactivity was captured by the drain system than was accounted for by Unit 1 SFP leak calculations. Evidence from the hydrogeologic site investigation confirms the source of this additional tritium radioactivity is from the Unit 2 SFP. Based on this understanding, additional NRC analysis used historical Unit 1 building foundation drain system hydrogen-3 (tritium) sample results to attempt to assess the age and variation of the Unit 2 SFP leak since 1999.

b. Findings and Assessment

No findings of significance were identified.

A review of daily boron concentration measurements in the Unit 2 spent fuel pool since the last refueling outage indicated a decrease of 7 parts per million (ppm) (normally 2,300 ppm) over a one year time period. This measurement provided a bounding water loss value of 2.6 gallons per day (gpd), with a large uncertainty of +/- 7.2 gpd. This uncertainty indicates that no definitive loss of spent fuel pool inventory could actually be determined with any certainty.

The licensee has pursued consistent efforts to inspect the Unit 2 spent fuel pool stainless steel liner for evidence of leaks. Approximately 40% of the liner was inspected by underwater video camera. No leakage was determined on the surfaces examined. The remainder of the pool liner surfaces is inaccessible to optical examination due to limitations imposed by the proximity of the fuel racks and other obstructions. Beginning in July 2007, Entergy lowered the water level in the Unit 2 fuel transfer canal, which is immediately adjacent to the spent fuel pool, in order to examine those surfaces for possible leaks. One pinhole leak was discovered and was subsequently repaired on December 15, 2007. An expert review of the material condition of the leak determined that it was due to an original welding construction flaw, and that there were no indications of any active corrosion on the transfer canal surfaces.

Notwithstanding that all identified potential leak locations have been repaired, most of the spent fuel pool surfaces remain unexamined, with the potential for unidentified leaks remaining. Since the Unit 2 spent fuel pool was constructed without a leak collection system, groundwater monitoring remains the only means for assessing leakage from the Unit 2 spent fuel pool.

.5B Unit 1 SFP Leakage

a. Inspection Scope

A review of available licensee records was conducted to search for any possible indications of the beginning or duration of the Unit 1 SFP leak. Records were also reviewed to evaluate the licensee’s response to the initial discovery of Unit 1 SFP leakage, and the adequacy of corrective actions to repair or mitigate the effects of the identified leakage based on regulatory requirements and information known at the time.

b. Findings and Assessment

No findings of significance were identified.

A search for historical Unit 1 control room logs and for Unit 1 spent fuel pool inventory makeup records was initiated, but no pre-1994 records were found. Without those records, which are no longer required to be maintained, no data was available to indicate past water inventory makeup trends. The water makeup records and control room log entries represented the only potential data records to evaluate the onset of Unit 1 SFP leakage, which remains indeterminate.

The initial licensee’s corrective action program identification and investigation of the leaking Unit 1 SFP (SAO-132 Report 94-06), identified a net fuel pool leak rate (subtracting evaporative losses) of 25 gallons per day, or 10 drops per second, attributed to age-related degradation of the fuel pool epoxy coating, which resulted in pool water penetrating through the fuel pool concrete walls and floors. The corrective actions associated with Report 94-06, included a large scope of investigative activities aimed at identifying potential leakage paths within the Unit 1 plant structures, including groundwater collected in the external Unit 1 building foundation drain system (Figure 2). Bounding dose calculations performed by the licensee in 1994, which assumed four times the identified leak rate released to the Hudson River, indicated that the resulting dose from such a liquid release would be <0.1% of the liquid effluent regulatory specification and ALARA guidelines.

The NRC conducted three separate team inspections in 1994 (specified in Attachment 1) to assess the licensee’s identification and resolution of the leaking Unit 1 spent fuel pool condition and based on a comprehensive review concluded that the licensee’s investigation was responsive to this concern and the potential impact on the public health and environment. Further, that the licensee’s investigation incorporated all reasonable probable pathways of release and had demonstrated no off-site dose impacts would be attributable to pool leakage based on enhanced environmental surveillance.

Entergy's investigative activities did not result in correcting the degraded condition of the Unit 1 spent fuel pools or otherwise eliminate the identified leakage. Unit 1 licensing and procedural requirements were reviewed and no corrective action program violations were identified. NRC requires safety-related functions of plant components to be repaired or corrected in accordance with 10 CFR 50, Appendix B, Criterion XVI. However, the leak rate from the pool did not affect the safety-related function of the Unit 1 spent fuel pool (associated with spent fuel cooling), and the off-site dose consequence of the leakage was evaluated and determined to have no significant dose impact. Therefore, there was no condition adverse to quality and no violation of NRC requirements identified.

This 1992 investigation was the earliest documentation confirming leakage of the Unit 1 SFP. Since 1992, the leakage rate remained constant until the Fall of 2005, when the Unit 1 West SFP was flooded up to allow fuel inspection as part of the future dry cask storage relocation of the spent fuel. After lowering the water level back down and draining the surrounding pools in November 2005, the Unit 1 West SFP leak rate increased to 70 gallons per day due to a higher water pressure forcing more water to drain through the preexisting cracks to the surrounding now drained Unit 1 spent fuel pools. Based on the tritium concentration measured in the Unit 1 West SFP and the current leakage rate, a comparison of tritium leaking from the Unit 1 West SFP and the total tritium collected by the Unit 1 building foundation drain systems could be compared. Latest calculations indicates that there is approximately three times more tritium collected than can be accounted for from Unit 1 West SFP leakage.[2]

Based on the hydrogeologic site investigation, it is now known that the source of the additional tritium activity is due to migration of tritium contaminated water from the Unit 2 SFP, in the unsaturated zone southward towards Unit 1 and being drawn into the groundwater cone of depression created by the Unit 1 building foundation drain system. Recognizing that the Unit 1 West SFP leak condition was stable at about 25 gpd prior to the Fall of 2005 with a stable radioactive source term, historical review of licensee data was used to evaluate the change in the Unit 2 SFP leakage over time since approximately 75% of the tritium collected in the Unit 1 foundation drainage system was due to the Unit 2 SFP leak.

This evaluation was considered necessary to help investigate the results of a sample taken in the Spring of 2000 from Monitoring Well-111 when Entergy was exploring the possibility of purchasing Unit 2. No tritium was detected in the sample. The monitoring well is located in the current Unit 2 SFP tritium plume. The sensitivity of the sample method should have detected any tritium above 270 pCi/L. This fact would indicate that the Unit 2 SFP tritium plume did not exist in the Spring of 2000, and that the SFP leak may have begun more recently. Entergy's site characterization report indicates the sample was not a reliable groundwater sample as it was taken from the surface of the well without any purging and was, therefore, not considered representative of the groundwater at this location. In order to determine the efficacy of the Spring 2000

[2] The March 16, 2006 Special Inspection Report indicated a higher unaccounted for tritium balance due to a calibration issue with a flow rate monitor, a condition that has been corrected.

Monitoring Well-111 sample and the possibility of a more recent SFP leak, the Unit 1 building foundation drain collection data was accessed to provide an indication of excess tritium infiltration (attributable to Unit 2 SFP leakage) around the time of the Spring 2000 Monitoring Well-111 sample compared to the present time.

If there was no tritium plume emanating from the Unit 2 SFP at that time, then there should be a significant reduction (approximately 75%) in the tritium input to the Unit 1 building foundation drain system. Otherwise, Entergy's site characterization model, which suggests a long-term tritium leak, would be reasonable. The following table summarizes data extracted by the NRC from licensee data. The two Unit 1 building foundation groundwater drain systems consist of the north curtain drain (NCD) and the sphere foundation drain (SFD). The combination of both of these two french drain type systems represents the total tritium collected annually based on weekly sample collections.

Unit 1 Drain Tritium Collection

Year	SFD uCi	SFD flowrate gpm	NCD uCi	NCD flowrate gpm	Total uCi	Total flowrate gpm	Corrected[3] uCi
1999	8.82E4	18	6.0E5	3	6.9E5	21	4.6E4
2005	2.67E4	24	5.8E4	3.6	8.5E4	28	5.6E4
2006	5.2E4	17	4.7E4	4	9.9E4	22	6.6E4
2007	2.6E4	11	2.7E4	2.8	5.3E4	14	5.3E4

As can be seen, in the final corrected column in the table above, there has been a consistent amount of tritium collection in the Unit 1 drain system that predates the "due diligence" sampling of Monitoring Well-111 in the Spring of 2000. This would indicate that the Unit 2 SFP tritium plume was being captured by the Unit 1 drain system in 1999 as currently characterized, and that the Spring 2000 Monitoring Well-111 sample may not be a valid sample. This confirms the designation as an invalid sample as stated in Entergy's hydrogeological final report.

Considering factors including the radiological and non-radiological contamination condition at Unit 1, Entergy determined that any immediate remediation (such as groundwater pump down) of the existing contaminated groundwater in the vicinity of the Unit 2 spent fuel pool would be inappropriate at this time. Such remedial action could adversely affect the current groundwater contamination condition, in particular, it would create a situation in which contaminated water that is currently collected, monitored and discharged from the Unit 1 drain systems in accordance with NRC regulatory requirements, to spread elsewhere unnecessarily. Accordingly, the NRC agrees that, in the absence of any over-riding public health and safety concern, pump and treat remediation of the Unit 2 SFP could adversely affect the spread of the Unit 1 groundwater contamination plume and is not advisable.

[3] In 2006, the SFD flowrate monitor was found to be significantly overestimating the flow rate by 50%; therefore assuming relatively constant annual groundwater flow, the total tritium results for the prior years was reduced by 50% to provide a normalized comparison.

.6 Hydrogeologic Investigations

a. Inspection Scope

NRC Region I Inspectors, and scientists from the U.S. Geological Survey (USGS) and NRC's Office of Research made numerous visits to the IPEC site to observe site features, test hole drilling and sampling, rock cores recovered from the test wells, groundwater quality sampling, tracer and pump test procedures, and other site characterization and monitoring activities. During these site visits, the inspection team interviewed Entergy staff and contractors, i.e., GZA GeoEnvironmental, Inc. (GZA) geotechnical engineers, geologists, and hydrogeologists, and examined their methods, analytical results and bases for conclusions regarding groundwater contamination transport at Indian Point Energy Center.

b. Findings and Assessment

No findings of significance were identified.

The purpose of the hydrogeological investigation was to identify the on-site, and potential off-site, pathways for the abnormal releases, and to define the conceptual site hydrologic model controlling the subsurface transport of the released radionuclides.

Initially there were significant uncertainties in defining the tritium pathway (the first detected abnormal release radionuclide). In discussions with GZA, it was apparent that the tritium source(s) and pathway(s) were not fully defined. Questions were raised as to the groundwater flow direction, which the IPEC FSAR Section 2.5 references indicated was to the south. Based upon water-level data taken by GZA from a series of installed test wells, the groundwater gradient was initially determined to be west to the Hudson River in the vicinity of the Screen Wall Structure building (near Monitoring Well-67). Upon close examination of the water-level data for the full complement of test wells, the groundwater flow direction was confirmed to be the west and, therefore, the tritium plume was determined to follow the gradient to the Hudson River. Tritium moves at the same rate as the groundwater since it is part of the molecular water composition. Analysis of monitored water levels, temperature and water quality demonstrated tidal effects from the river affecting groundwater flow conditions along the river bank and upgradient to the Discharge Canal.

The question of preferential flow pathways was raised due to the nature of the bedrock underlying the IPEC site, the Inwood Marble, being a metamorphosed carbonate with numerous fractures. These fractures, which can be observed on-site and in the Verplanck Quarry as shown in Figure 3, were inspected for the possibility of solutioning and connectivity. The rock cores collected during the drilling of the test wells were examined for fractures, solutioning and fracture filling. In order to confirm the Entergy/GZA determinations a range of possible conceptual site models were examined to determine the influence of fracturing, solutioning and fracture filling on contaminant transport. In order to fully investigate and independently analyze alternative conceptual site models involving preferential groundwater flow pathways, NRC developed an

Interagency Agreement with the USGS - New York Water Science Center located in Troy, New York.

The USGS conducted a detailed flow-log analysis for hydraulic characterization of selected test wells. This analysis examined fracture geometries and hydraulic properties in the bedrock using flow logs, as well as downhole caliper, optical- and acoustic-televiewer, and fluid resistivity and temperature logs, collected in the test wells by Geophysical Applications, Inc. under the direction of GZA. The USGS analysis determined the distribution and character of fracture-flow zones. Hydraulically active fractures were identified in these zones. Transmissivity and hydraulic heads in these flow zones were estimated using the flow-log analysis method. As reported in USGS Open File Report 2008-1123 "Flow-Log Analysis of Hydraulic Characterization of Selected Test Wells at the Indian Point Energy Center (IPEC), Buchanan, New York" (ADAMS Accession No. ML081120119), the flow-log analysis was corroborated with pump test and tracer test results from GZA's site characterization and analyses.

Figure 4 shows the presence of intersecting (conjugate) fracture sets which provide higher permeability zones and create directional flow properties (anisotropy). These analyses were confirmed by pump test results, and later, tracer test results and observations showing distinct fracture zones and variable permeability in the Inwood Marble between the Unit 1 and 2 SFPs extending west to the Discharge Canal. No solution features affecting radionuclide transport were observed or detected by the field testing and USGS independent analysis. However, fracture connectivity was observed and is a contributor to preferential flow and transport, particularly in partially-saturated bedrock (i.e., above the water table) as demonstrated by the GZA tracer test results. Certain site areas subject to extensive rock backfills, such as the excavated-blast depressions in the transformer yard and along the river, which are porous-flow dominated rather than fracture-flow dominated as indicated in the bedrock.

Early in the investigations, the Discharge Canal was thought to capture the tritium plume. NRC staff questioned this assumption and encouraged its testing. GZA installed Monitoring Well-37 west of the Canal and down gradient of the plume to test the assumption. Sampling in Monitoring Well-37 confirmed that the tritium plume did continue west under the canal toward the Hudson River; however, a significant amount (perhaps up to 50%) of tritium was captured by the canal. Sampling in Monitoring Well-37 also identified strontium-90 which extended the scope of the investigation.

As the conceptual site model (CSM) was developed using observed tritium and strontium-90 monitored data from the numerous monitoring wells, the role of backfill material around buildings and in excavated depressions (e.g., transformer yard and along the river) was investigated by GZA. The role of storm drains, sump pumps and curtain drains on the local hydrology was also investigated and analyzed. The conceptual site model, as reported in the licensee's Hydrogeological Site Investigation Final Report (GZA report), recognized the affect of these features relative to the observed tracer test results and contaminant plume behavior. The conceptual site model incorporated both natural features (e.g., water-levels and flow directions) and human-made features (e.g. building foundations, backfills, curtain drains, storm runoff drains and manholes). The conceptual site model considered percolation to the

unsaturated zone, where the Unit 2 tritium source emanates, and flows to the water table. The strontium source was determined to enter the water-table via the north curtain drain surrounding the Unit 1 SFP, and also from the spray foundation sump. Both the tritium and strontium plumes migrate through the connected fractured zones to the Hudson River. Cross-sectional diagrams from the GZA report, shown in Figure 5, depict the flow and transport
pathways to the river, including the location of monitoring wells down gradient of the radionuclide sources. Tracer test and radionuclide sampling data from these monitoring wells support the conceptual site model assumptions.

A pump test using Recovery Well-1, with observations in the surrounding monitoring wells, was performed to test the feasibility of a pump, monitor and discharge remediation approach for the tritium plume, and to create a depressed water table (drawdown cone) beneath Unit 2 SFP to capture and provide early detection of abnormal releases. The operation of the Recovery Well-1 caused cesium-137, which had not been previously detected in monitoring wells, to migrate to Monitoring Well-31 and Monitoring Well-32 (west of the Unit 1 and 2 SFP's). This test confirmed the presence of cesium-137 in the fractured rock, and the connectivity of the fractures in the aforementioned fracture zones between the Unit 2 and 1 SFP's. The migration of cesium-137 from Unit 1 to Unit 2 during the test confirmed that the pump test should be conducted at very low pumping rates in the event that other radionuclides were present in the fractured rock and could become mobilized. The fracture filling in the bedrock appears to adsorb the cesium during ambient groundwater flow conditions.

Using insights from this pump test, GZA planned and conducted a tracer test adjacent to Unit 2 SFP at the base of the construction pit where the original abnormal releases of radionuclides were observed. A fluorescein dye tracer was introduced in a shallow borehole above the water table. At the suggestion of NRC staff, the tracer sampling continued for a significantly longer period of time than would be normal to fully detect and analyze the transport pathways. The tracer results confirmed the aforementioned conceptual site model pathways, and identified the role of the fractures in creating preferential transport in the unsaturated zone, and the role of human-made features relative to the observed tritium concentrations in the monitoring wells and Manhole 5 adjacent to Unit 2 SFP. The tracer sampling identified the contaminant pathway direction, transport rate and attenuation for both the tritium and strontium plumes. Since strontium-90 is adsorbed by the fracture filling materials (e.g., clays), the tracer moved at a faster rate than the strontium plume. The residual cesium-137 appears to be relatively immobile due to adsorption and the relatively slow groundwater velocity in the fracture zones until increased by local flow perturbations such as groundwater pumping.

The extensive IPEC site characterization data as reported in the GZA report includes: water levels; tidal effects; upward and downward flow components determined by flow meters and by using the Waterloo packers (i.e. inflatable bladders to vertically isolate fracture zones in a well); tritium and strontium concentrations; and pump and tracer test results. This database provides valuable site-specific information to confirm the conceptual site model (CSM) and dose calculations. This information also provides a valuable two-year baseline for future long-term monitoring and re-evaluation of the conceptual site model since seasonal groundwater flow dynamics, episodic recharge and potential future releases may alter the assumptions in the CSM. This information is also critical in determining the adequacy of the Entergy's chosen remediation approach of monitored natural attenuation for the tritium and strontium-90 plumes.

Monitored natural attenuation refers to the natural groundwater removal of residual contaminants after the source of contamination has been secured, and the radioactive decay acts to diminish the remaining residual radioactivity. Monitored natural attenuation requires the elimination of the contaminant sources, detailed monitoring of the plumes' behavior through a confirmatory groundwater monitoring program and confirmation of the conceptual site model, over time.
The licensee indicated that its long-term groundwater monitoring program will incorporate monitored natural attenuation and have a detection capability for potential future abnormal releases. Future NRC inspection will review the program details to focus on achieving the goals of monitored natural attenuation and detecting future leaks. Specific areas of review include determining which monitoring wells and what monitoring frequencies are needed to demonstrate monitored natural attenuation, early radionuclide leak detection and if the assumptions in the conceptual site model are valid. The long-term groundwater monitoring program will be reviewed in a future NRC inspection to ensure there is sufficient detection sensitivity and monitoring frequency to detect changes in Unit 2 SFP leakage and the capability to detect leaks from other plant components in the presence of existing groundwater contamination.

.7 Prior Indications of On-site Groundwater Tritium Contamination

a. Inspection Scope

The inspectors reviewed NRC required documentation affecting the identification of potential and actual leaks of radioactivity outside of plant systems. The records were reviewed to identify any historical survey data that the licensee possessed that would indicate prior knowledge of any groundwater contamination issue that was not evaluated as required. Title 10 CFR 50.75(g) requires records to be retained of past on-site contamination spills. These records for the Indian Point site were reviewed for relevance to the current site condition.

NRC IE Bulletin No. 80-10, ΑContamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to Environment@ requires licensees to review their facility design and operations to identify nonradioactive systems, that could become radioactive through interfaces with radioactive systems, to include leaks and valve misalignments. The Bulletin required routine sampling and analysis for the identified nonradioactive plant systems be established in order to identify
any contaminating events that could lead to unmonitored, uncontrolled releases to the environment. In response to the Bulletin, the licensee developed lists of affected plant systems and sampling periods. The inspectors also reviewed the licensee's program for the sampling of on-site storm drain systems for radioactive liquids and sediments. Also, the inspectors reviewed the results of the "due diligence" sampling that was conducted in early 2000 to identify outside plant areas with residual contamination. These results were also screened for potential evidence of the preexisting groundwater contamination

condition.

b. Findings and Assessment

No findings of significance were identified.

The 10 CFR 50.75(g) decommissioning file included records of the prior Unit 2 SFP leak from October 1, 1990 – June 9, 1992 as documented in corrective action report (SAO-132, 92-08). These records indicate an effective cause determination and repair of the condition. In addition all affected soil was excavated to a depth of eight feet and the affected 35 cubic yards of soil was shipped off-site as radioactive waste, with no residual soil contamination remaining. No evidence of groundwater contamination was determined.

The Unit 1 SFP leak assessment corrective action report (SAO 132 94-06) and hydrology report (Whitman 1994) were included in the decommissioning file, identifying that most of the 25 gpd leak identified in 1992 would be intercepted by the Unit 1 building foundation drain system. Any portion not intercepted by the drain system would likely follow a shallow ground water flow pathway into a small stream discharging into the Hudson River some 1700 feet southwest of Unit 1. Based on this information, the licensee added environmental sampling stations to include the small stream south of Indian Point as well as the Trap Rock Quarry (0.7 miles south of the plant) and an unused groundwater well located off of Fifth Street in the town of Verplanck (1.3 miles south of Indian Point). Environmental records of those sampling activities did not identify any radioactivity in these samples that was plant-related.

Decommissioning file records of the Unit 2 SFP leak that was discovered in September 2005, includes records indicating a 2.6 gpd bounding leak rate was determined in a November 21, 2005, boron-loss mass balance calculation. The current hydrogeologic site investigation report completes the groundwater contamination records in the 10 CFR 50.75(g) decommissioning file.

Other miscellaneous documents were reviewed including some legacy records of low level Cs-137 contamination found in, and associated with, Unit 1 storm drain lines (1-50 picocuries per gram) that predated commercial operation of Units 2 and 3. One area, 10 feet X 70 feet X 3 feet deep, identified in July 1990 on the north side of the Unit 3 fuel storage building, was originally excavated storm drain material with residual levels of Cs-137 (30 pCi/g) from Unit 1 operations; it was later paved over. This action included a dose evaluation which indicated the area would result in much less than 1mrem/yr, which
would not require immediate cleanup in accordance with NRC site cleanup screening level of 5 mrem/yr (NUREG/CR-5849).

Review of the “due diligence” site assessment conducted by Canberra Services on February 14 - 22, 2000, identified various areas inside the restricted area with detectable radioactivity. Several monitoring wells were installed and sampled. None of the groundwater samples indicated any detectable plant-related radioactivity.

The IE Bulletin 80-10 program specific to on-site storm drain monitoring was fairly extensive and provided detailed records since 1981. Review of the site wide storm drain system data did not indicate a history of the current extent of elevated tritium contamination. No historical marker was indicated in the storm drain sample data as to when the tritium leaks may have been initiated.

Entergy's IE Bulletin 80-10 program ("IPEC Storm Drain Sampling Procedure", O-CY-151-, Rev. 3) has been recently revised, consolidating two previously separate Unit-specific programs with an updated map of the Unit 1, 2 and 3 storm drain systems, and incorporating a consolidated sampling schedule, with appropriate frequencies, that includes monthly sampling for sensitive storm drain outfalls. The improved program now includes specific sample detection criteria requiring management involvement.

.8 Remediation and Long Term Monitoring Plans

a. Inspection Scope

In addition to providing the hydrogeologic site investigation final report to the NRC on January 14, 2008, a subsequent Memorandum dated January 25, 2008 (ADAMS Accession No. ML, 080290204) provided a synopsis of the Long Term Monitoring Plan Bases. These documents were reviewed along with a number of Entergy and GZA implementing procedures that provide a framework for addressing the current and future groundwater contamination issue. Several meetings were also held between the NRC, USGS and NYS DEC in January and February 2008 to discuss the adequacy of Entergy's plans and procedures.

b. Findings and Assessment

No findings of significance were identified.

Based on the installation of on-site monitoring wells, 36 out of 39 monitoring wells were selected by Entergy for continued sampling at established frequencies. In addition, three storm drain manholes were included in the sampling plan to monitor drainage from the Unit 2 containment footer drain and the Unit 3 foundation and containment footer drains. This initial sampling program consists of 378 annual samples to provide trending information on the current contaminant plumes and provide for early detection of leakage from other potential on-site sources to comply with the requirements of NEI 07-07, "Industry Ground Water Protection Initiative", for early detection and reporting of on-site spills or inadvertent contamination of groundwater.

In addition, the on-site storm drain system for Units 1, 2 and 3 was visually inspected using remote camera technology and large volumes of material (over 100 tons) were removed to complete the inspection and make requisite repairs. During NRC inspection of prior sampling evidence of groundwater contamination, in the March 16, 2006, special inspection report, the storm drain sampling program was assessed as a segregated program (between the operating Units) without proper program administration or data trending review. Since those observations, Entergy has renovated the storm drain systems, validated their connections and flow directions, and consolidated the program into one site-wide program with individual sample detection criteria that initiates management review. The current storm drain sampling program requires over 140 samples per year to detect potentially leaking plant systems as part of the IE Bulletin 80-10 requirement.

Currently, there is no periodic trending review of storm drain sampling data or use of this program with the groundwater monitoring program. Since one of the main functions of storm drains is to remove surface runoff water, many of the storm drains included in the sampling program may not provide any indication of below ground leaking plant systems or components. Since the site groundwater investigation has established the water table and groundwater gradients, the licensee has initiated actions to evaluate the storm drain systems for additional input to the long-term monitoring program.

The long term monitoring plan implementing procedures incorporate periodic sampling from a groundwater monitoring network composed of 36 monitoring wells and numerous other sampling locations. The current groundwater plumes are mapped spatially among this network of monitoring wells to allow future monitoring of the plume's footprint. At the conclusion of this inspection, the licensee was still in the process of defining and establishing the parameters of its long-term monitoring program.

Early in the Unit 2 spent fuel pool leak investigation, Entergy reviewed detailed fuel pool boron sampling data in an effort to determine net leakage losses from the fuel pool, since boron loss would not be affected by pool evaporative losses and any reduction in boron concentration would be due to pool leakage. Transfers of spent fuel and reactor water during refueling outages set a new boron solution level and trends of boron concentration losses after each refueling outage. This trending of boron data provided an initial Unit 2 SFP loss rate of approximately 2.6 gallons per day (approximately 1 drop per second) calculated by Entergy in September 2005. Although there are some complicating factors (e.g., variance in boron data measurement and any unidentified fuel pool cooling system leaks), this approach does provide an early indication of net change in spent fuel pool leakage.

Entergy plans on removing the spent fuel and draining the Unit 1 spent fuel pools by the end of 2008. Some water may remain in the bottom of the pool to reduce the possibility of airborne contamination and provide shielding of remaining sludge. Sludge removal is expected to be completed in early 2009. After completion of these activities, the source of the Unit 1 plume will be eliminated allowing residual radioactivity removal through continued purging from the Unit 1 building foundation drain system and through natural attenuation processes.   Relative to Unit 2, the licensee has taken action to repair all identified liner leak imperfections, and has identified a program for monitored natural attenuation on the presumption that leakage has been terminated, based on its current assessment of groundwater tritium concentrations. However, neither the licensee nor the NRC is conclusive at this time, since only 40% of the liner surface was accessible for inspection; and it is too early to detect any significant decline in tritium concentrations (with respect to the natural variability in groundwater flow). Notwithstanding, it is expected that the licensee's implementation of its long-term monitoring program will establish sufficient data to permit a conclusive determination in the near term.

The current dose significance of the Unit 2 SFP tritium leak rate is 1000 times lower than the current Unit 1 plume (approximately 0.000002 mrem/yr versus 0.002 mrem/year), and therefore, additional actions beyond long-term groundwater monitoring of both groundwater plumes by Entergy are not warranted and the current approach is acceptable to the NRC.

Further definition of the long term monitoring plan and licensee commitment to this groundwater surveillance program will be pursued through continuing inspection activities in 2008. These future inspection activities will verify completion of Entergy's planned remediation activities, and to review plume attenuation results to confirm Entergy's site groundwater characterization conclusions.

.9 Regulatory Requirements

a. Inspection Scope

The following regulations were reviewed to identify any areas of noncompliance.

The NRC regulates the radioactive effluent releases from nuclear power plants through guidelines based on instantaneous maximum concentration values specific for each radionuclide as well as regulatory limits on potential doses to the public. The release limits are based on 100 mrem total effective dose equivalent per year. In addition, licensee's are required to meet the ALARA design objective guidelines of 3 mrem to the total body per reactor and 10 mrem to the maximum organ dose receptor per reactor (10CFR50, Appendix I). There are also total site annual exposure limits to actual members of the public from all pathways of 25 mrem to the whole body, 75 mrem to the thyroid and 25 mrem to any other organ (40CFR190.10(a)).

Effluent releases are reported by each nuclear power plant licensee to the NRC on an annual basis with calculated maximum doses to the public and comparison to the above indicated NRC limits. In addition, to provide a verification of these calculated releases, a radiological environmental monitoring program is conducted by the licensee providing off-site environmental sample measurement results for biologically sensitive pathways of exposure to man especially in locations directly downstream or downwind of the nuclear power plant. Spills or leaks on the site property are required to be recorded to support future decommissioning activities (10CFR50.75(g)).

Unless drinking water is provided from on-site groundwater wells, the environmental monitoring program does not require on-site groundwater monitoring. This area of the regulations is currently under review. The industry has adopted a Groundwater Protection Initiative (Nuclear Energy Institute; NEI 07-07, August 2007) to initiate on-site groundwater monitoring at all nuclear power plants, and the NRC is proposing additional rulemaking and guidance (10 CFR 20.1406 and Regulatory Guide 4.21) to address the potential for leaks into the groundwater and the need to monitor this potential effluent pathway.

b. Findings and Assessment

No findings of significance were identified.

Instantaneous release rates are limited by procedures that establish gaseous and liquid release radiation monitor system setpoints and automatic discharge valve closures. Based on review of monitoring well sample results from October 2005 through

December 2007, groundwater effluent instantaneous release concentrations were always a small fraction of the regulatory limits.

The annual and quarterly liquid effluent public doses were calculated annually for 2005 and quarterly and annually for 2006 based on a rain precipitation water infiltration drainage model developed by Entergy's hydrogeologists to derive groundwater flux values to drive the contamination concentrations obtained from monitoring well sample results. In 2005, when few samples were available, the maximum monitoring well sample results were used in the calculations. For the quarterly 2006 groundwater effluent calculations, when multiple sample results were available, the monitoring well sample results were ranked (low to high) and the 75th percentile values were used to derive a best estimate of the groundwater releases to the Hudson River. A half-tidal surge of the Hudson River was used as a final dilution of these releases and dose calculations were performed based on the Indian Point Energy Center Off-site Dose Calculation Manual (ODCM) methodology. The ODCM incorporates exposure pathway dose calculations based on Regulatory Guide 1.109. Doses were calculated based on Hudson River specific bioaccumulation of contaminants in fish flesh and based on infant, child, teen and adult fish consumption rates. Various organs concentrate various radionuclides at differing rates, so doses are calculated for bone, liver, total body, thyroid, kidney, lungs, and gastrointestinal tract, based on applicable dose factors for each critical organ. The maximum age group and organ is reported.

For 2005 and 2006, the following doses were reported for both normal and groundwater liquid effluents.

2005 Liquid Effluents	Units 1 & 2 (mrem)	Unit 3 (mrem)	Limit (mrem)	Max % of Limit
Routine max quarter	2.93E-4 TB[4] 4.68E-4 O[5]	3.29E-4 TB 3.85E-4 O	1.5 5	0.02 0.009
Routine annual	8.11E-4 TB 1.31E-3 O	4.45E-4 TB 5.4E-4 O	3 10	0.098 TB[6] 0.11 O[6]
Groundwater annual	2.12E-3 TB 9.72E-3 O		3 10	0.07 0.1
2006 Liquid Effluents				
Routine max quarter	7.04E-4 TB 1.03E-3 O	6.8E-5 TB 7.6E-5 O	1.5 5	0.05 0.02
Routine annual	8.8E-4 TB 1.26E-3 O	1.27E-4 TB 1.6E-4 O	3 10	0.09 TB[6] 0.085 O[6]
Groundwater annual	1.78E-3 TB 7.21E-3 O		3 10	0.06 0.07

These maximum hypothetical doses represent approximately 0.1% of the ALARA design objectives for liquid effluents (3 mrem and 10 mrem per year per reactor) for Units 1 and 2, combined with the groundwater releases attributed to Units 1 and 2.

In conclusion, based on a review of applicable NRC radiation protection regulations, all effluent and environmental survey and reporting requirements have been met, indicating that the existing groundwater contamination conditions represent a small fraction of regulatory limits and no violation of these requirements have been identified.

[4] TB – Total Body exposure
[5] O – Maximum Organ exposure
[6] Represents total dose from Units 1&2 and groundwater

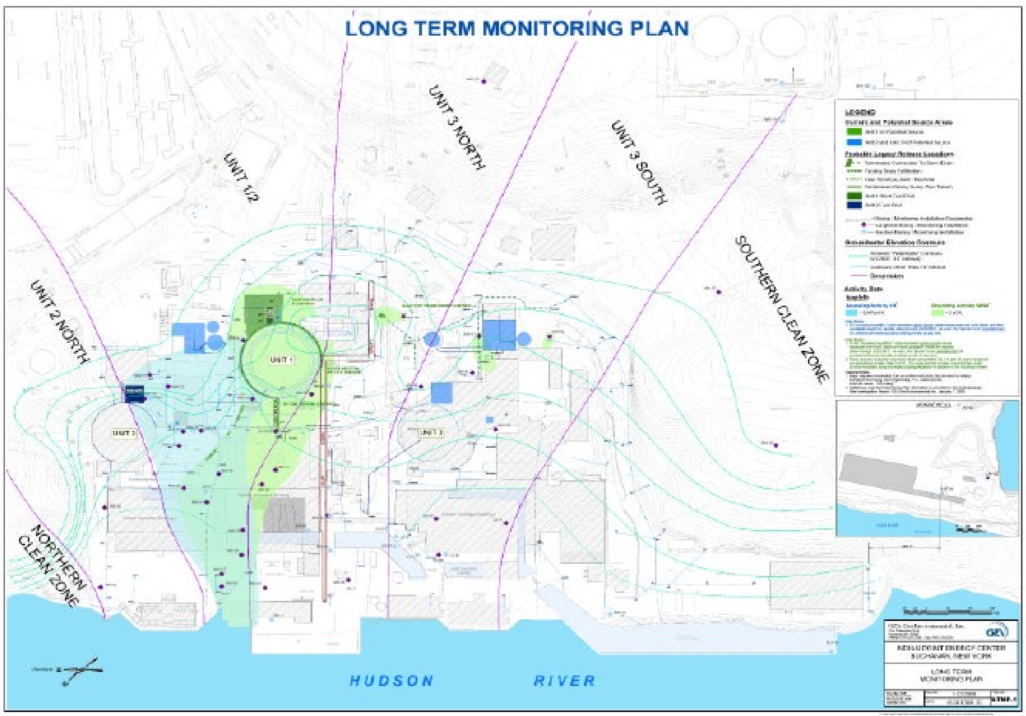
4OA6 Meetings, including Exit

.1 Exit Meeting Summary

The inspectors presented the Inspection results to Mr. D. Mayer and other licensee and New York State representatives on May 7, 2008. The licensee acknowledged the findings presented. Based upon discussions with the licensee, none of the information presented at the exit meeting and included in this report was considered proprietary.

Figure 1

Long Term Monitoring Plan

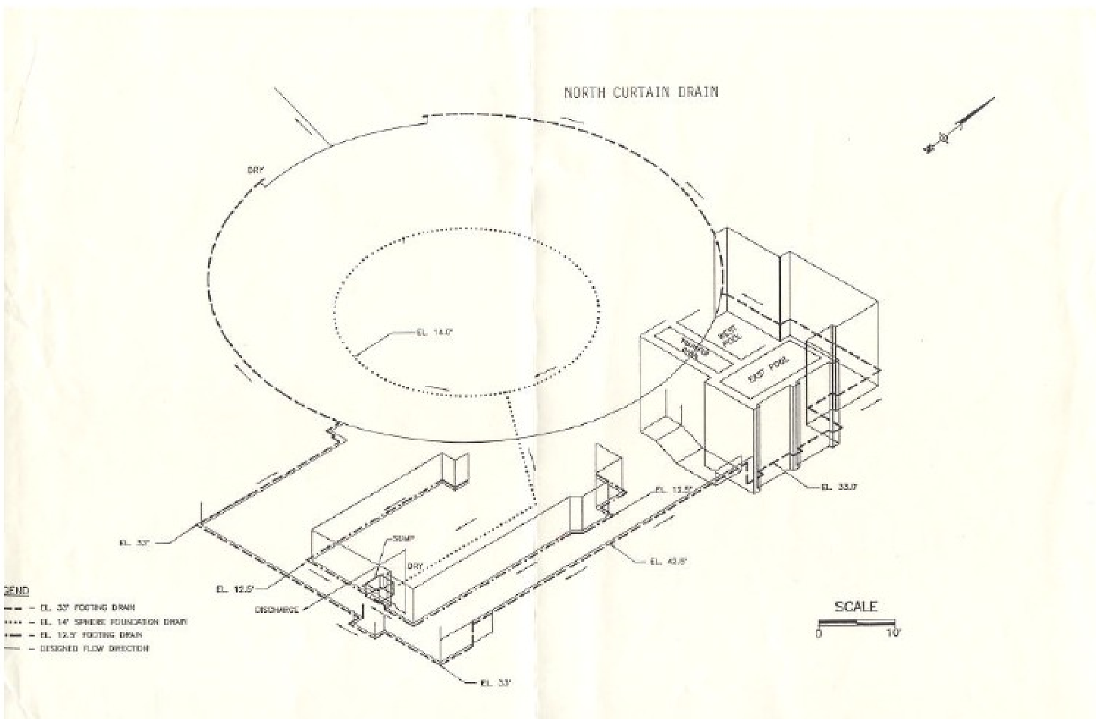


Detailed plume radionuclide concentration values at each monitoring well are provided in Attachment 2.

Enclosure

Figure 2

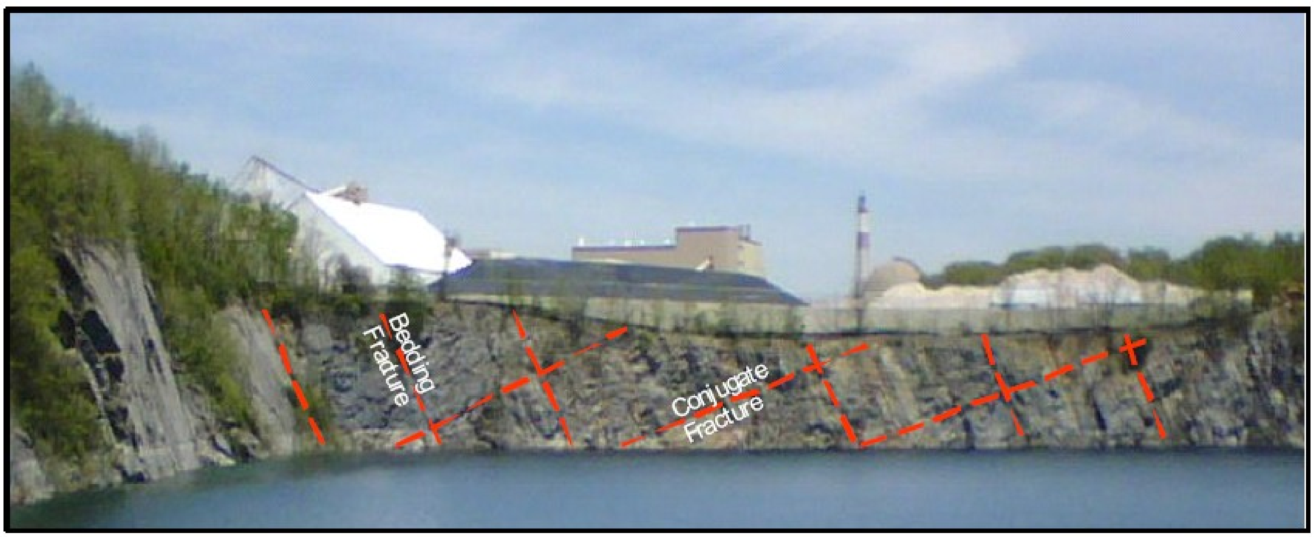
Unit 1 Building Foundation Drain System



Enclosure

Figure 3

Observed bedding and conjugate fractures in Verplanck Quarry (from USGS)



Enclosure

Figure 4

Downhole Flow Meter and Geophysical Survey
Example from Monitoring Well Monitoring Well-58

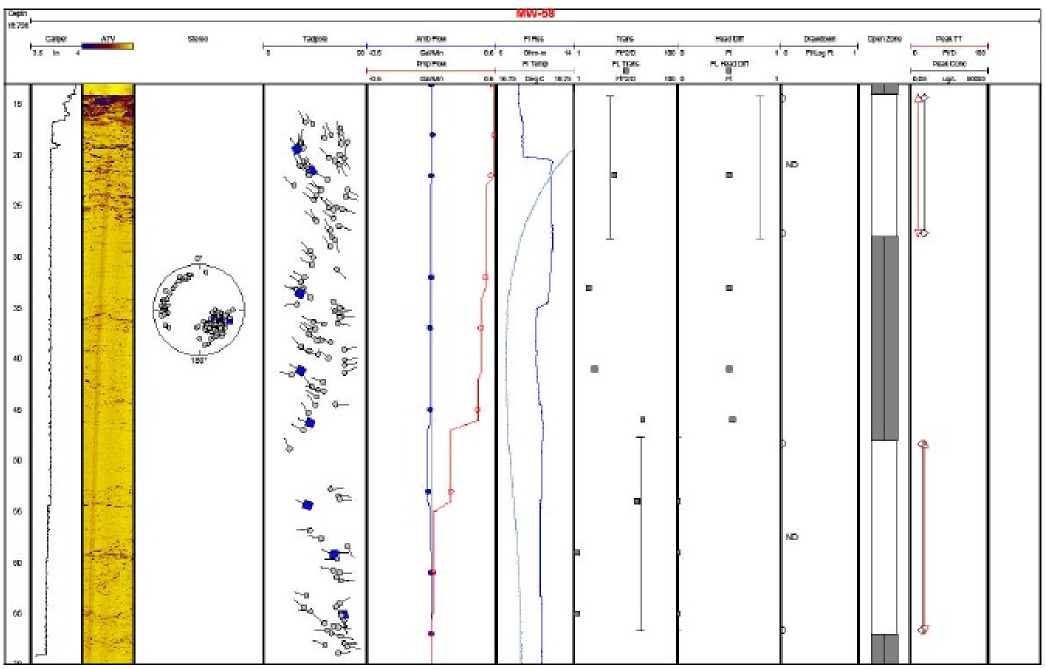
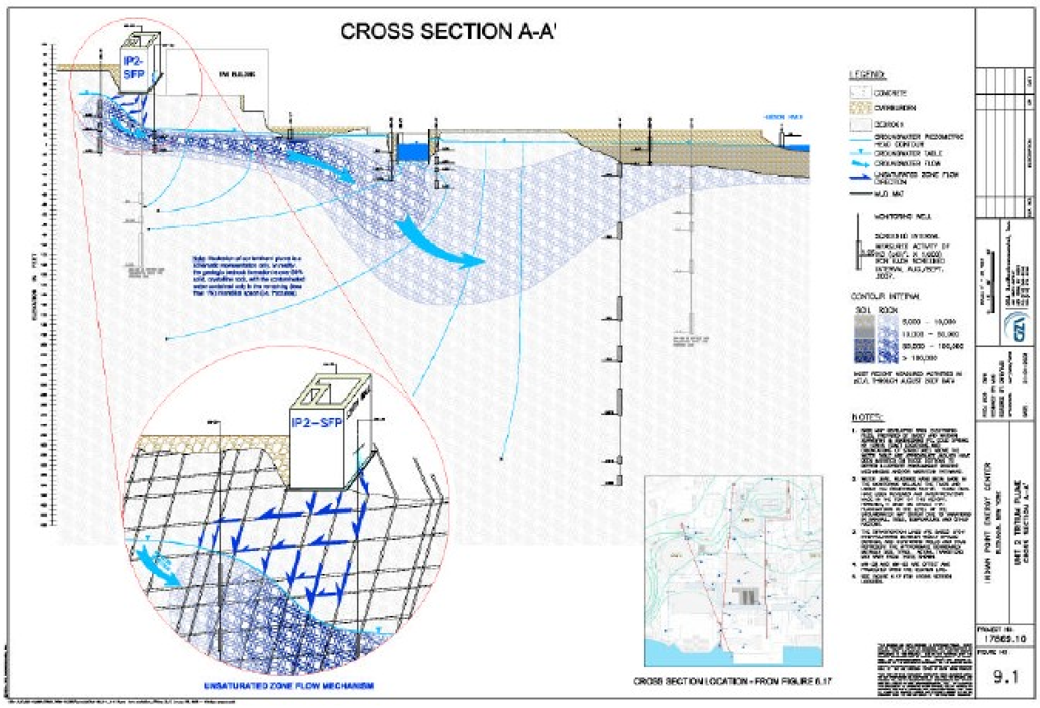


Figure 2. Analysis of geophysical and flow data to identify fracture zones and properties (from USGS)

Enclosure

Figure 5

Unit 2 Spent Fuel Pool Tritium Plume Cross Section



Enclosure

ATTACHMENT 1

Indian Point Contaminated Groundwater Investigation Time Line

<u>Date</u> <u>Event</u>

<u>Unit 1 Spent Fuel Pool Timeline</u>

Unit 1 ceased commercial operations on October 31, 1974

1. April 1990: A nuclear plant operator observed higher than usual frequency of fuel pool makeup than usual, initiated an investigation by Con Edison.

2. 1991: Con Edison began sampling the north curtain drain (NCD) and sphere foundation drain sump (SFDS) for tritium and established separate liquid discharge paths.

3. May 1992: Completed calculations of unaccounted water loss – 25 gpd leakage.

4. May 1994: A task force organization was created with a Unit 1 SFP Project Manager position reporting to the Plant General Manager. Individuals from Chemistry, Operations Maintenance, Health Physics and Engineering were represented.

5. May-June 1994: NRC inspection (Drs. Bores and Jang) to investigate Unit 1 SFP leakage (50-03/94-01) Boron concentration mass balance indicated 91 gpd leak rate to the SFDS and 1.5 gpd to the north curtain drain. Tritium concentration mass balance indicated 73 gpd to the SFDS and 1.2 gpd to the NCD. Hydrogeologist study indicated that the groundwater movement was about 10 ft/day and would flow towards the quarry, not the Hudson River. No violations were identified.

6. July 1994: Whitman hydrogeology report investigation of Unit 1 SFP leak migration concluded that "most" of the leakage would be captured by the Unit 1 building foundation drain system and the rest would migrate to the South in the shallow zone and could be detected in the creek bordering south of the plant and in the Trap Rock Quarry. These sample locations were added to the REMP program.

7. August 1994: NRC inspection (Bores/Jang) to review licensee's leak investigation (50-03/94-02). Hydrogeologist completed study indicated that groundwater at the site flowed upward and either west or south into the Hudson River. No violations were identified.

8. December 1994: NRC inspection (Bores, Jang, Erikson, Noggle) inspect compliance with Bulletin 94-01 (fuel pool potential siphoning), leak investigation, and SAFSTOR approval (50-3/94-80). Confirmation of tritium in the sphere foundation drain sump that drains groundwater from the bottom of the Chemical Systems Building of Unit 1 in May 1994, provided evidence that the Unit 1 SFP system was leaking beyond the plant structure and resulted in initiating a corrective action SAO-132 report (94-06). 10CFR50.59 evaluations between March 9, 1992 and December 1994 were reviewed and found to be complete and met requirements. In October 1994, boron concentration was increased in the SFP and fluoresce in dye tracer was

added to

the water storage pool to detect these sources in the NCD and SFDS. As of mid-December, no increased boron or indications of tracer were detected in either of these Unit 1 drains. Tracer did indicate that the SFDS had been discharging through a Unit 3 storm drain to the discharge canal. Con Edison subsequently rerouted this discharge by hard pipe through the Unit 1 River water system into the discharge canal. NCD was diverted to the Unit 1 sphere sump where this discharge was pumped to the liquid radwaste processing system. The on-site stream was added to REMP monitoring for tritium on a quarterly basis. No violations were identified.

9. January 2, 1996: SECY-96-01, Decommissioning Plan for SAFSTOR and amendment of license for Unit 1 was approved.

10. June-August 1996: NRC inspection (Jang) to review followup actions: modification to north curtain drain for recapture, new RMS detector installed in SFDS (50-3/96-04).

11. February-March 1998: NRC inspection (Jang) to review followup actions: effluent controls and trending of SFP inventory (50-3/98-02).

12. May-June 1998: NRC inspection (Ragland) reviewed schedule for draining and cleanout of pools (50-03/98-04). Con Edison removed all irradiated hardware from both the East and West Unit 1 SFPs.

13. November-December 1998: NRC inspection (Ragland) verified that irradiated hardware had been removed from the East pool and shipped off-site during May-August 1998, with the East pool ready for desludging and draining. PCBs detected in water storage pool sludge. (50-03/98-17).

14. December 1998-February 1999: NRC SAFSTOR inspection (Dimitriadis) (50-03/98-19). Work in progress in draining and desludging various pools. While desludging the water storage pool, PCBs were detected. Due to known leakage of this pool, the NCD was diverted into the Unit 1 sphere annulus for waste processing.

15. April-June 1999: NRC inspection (50-03/99-03) NRR reviewed a Unit 1 safety evaluation for modifications to the SFPs.

16. June-July 1999: NRC inspection (Ragland) reviewed monitoring of pool leakage, north curtain drain water was being treated by mechanical and charcoal filtration. Water storage pool cleanup in progress (50-03/99-06).

17. April 7, 2003: Unit 1 Remediation plan was approved to accomplish several objectives that included pursuing sealing the Unit 1 East SFP, transferring the spent fuel into that pool, and draining the leaking Unit 1 West SFP, thereby stopping the leak.

18. 2004: Insitu dry storage option was proposed by Unit 1 project team to stop the leak. Too many uncertainties surfaced regarding potential airborne radioactivity and future floodup effects on fuel integrity upon final spent fuel removal.

19. September 19-November 17, 2005: The Unit 1 West SFP was flooded up for spent fuel inspection for material condition evaluation. After drain down, Unit 1 SFP leak rate recalculated to be 70 gpd.

20. January 16, 2006: Unit 1 drain system collects seven times more tritium than can be attributed to the current 1 SFP leak rate.

21. March 21, 2006: NRC sample results of Monitoring Well-37 strontium-90 analyses were received indicating 26 pCi/L. This was the first indication that strontium-90 was likely being released in the groundwater to the Hudson River. Initial bounding calculations were revised, indicating less than 0.1% of effluent release limits.

22. April 17, 2006: Due to the 3/21/06 discovery of strontium-90 in Monitoring Well-111, the licensee initiated demineralization of the Unit 1 SFP 40 hrs per week in order to reduce leaking source term. Final assessment of Unit 1 SFP leakage calculations indicated 70 gpd post-drain down since November 2005.

23. April 24, 2006: Updated dose assessment based on 2/28/2006 methodology using more recent monitoring well data and maximum concentrations of hydrogen-3 (tritium), strontium-90 and nickel-63: 2.5E-3 mrem total body and 1.1E-2 mrem maximum organ (adult bone). Strontium-90 analysis was added to REMP fish, Hudson River and sediment samples.

24. August 9, 2006: After completing a temporary system modification, Entergy began continuous cleanup of the Unit 1 West SFP.

25. November 13-17, 2006: NRC on-site team inspection to review Unit 1 SFP leak history and hydrology results of a 3-day pump down test of Recovery Well-1.

26. April 2007: Revised calculation of tritium mass balance for Unit 1 SFP based on total radioactivity per year (based on 65 gpd leak rate) versus total radioactivity collected in the Unit 1 building drains for 2006. The Unit 1 SFP releases accounted for only 30% of the tritium collected in the Unit 1 drain system.

27. June 6-22, 2007: An expanded control zone fish split sampling exercise was conducted to include a second control location in the Catskills to help evaluate background levels of strontium-90 in fish.

Unit 2 Spent Fuel Pool Timeline

Operating license issued September 28, 1973

1. October 1, 1990: Unit 2 SFP stainless steel liner was perforated by a diver during re-rack cutting operation, but was not identified at that time.

2. May 7, 1992: Unit 2 SFP liner was discovered to be leaking (about 50 gpd), due to outside visible boric acid deposits on the wall of the fuel service building. Condition report determined cause and examined all other liner work areas for similar perforations. Entergy excavated 35 cubic yards of soil to a depth of 8 feet leaving no detectable contamination.

3. June 9, 1992: Under water epoxy temporary patch was installed, sealing the leak.

4. June 12, 1992: A steel box was welded over the liner perforation permanently sealing the leak completing corrective actions for this fuel pool leak event.

5. September 1, 2005: Initial discovery of the Unit 2 spent fuel pool leak. Contamination was first detected on a swipe sample of the exposed crack in the SFP south wall excavation area at approximately 65-foot elevation. The NRC resident inspector was informed.

6. September 12-15, 2005: NRC initial radiological scoping inspection and dose assessment, 0.00002 mrem/year based on 2 L/day leak rate.

7. September 20, 2005: NRC Special Inspection Charter was issued, followed by a press release announcing this action.

8. October 5, 2005: Tritium was discovered in the Unit 2 transformer yard Monitoring Well-111. This was the first location removed from the Unit 2 SFP indicating a groundwater contamination concern.

9. October 27, 2005: Unit 2 SFP liner inspection begins with underwater camera inspection to identify any leaks. Visual indications were followed by vacuum box testing.

10. October 31, 2005: NRC Executive Director for Operations issued Reactor Oversight Process deviation memorandum to provide additional NRC resources and continuing NRC inspection of the groundwater contamination investigation through 2006.

11. November 3, 2005: Licensee submitted a non-required 30-day report to the NRC, based on tritium results for Monitoring Well-111 (0.0002 uCi/ml) that were above the radiological environmental monitoring program (REMP) reporting criteria for non-drinking water samples (0.00003 uCi/ml). However, Monitoring Well-111 is an on-site well not representative of an off-site environmental sample therefore, no NRC report was required.

12. November 7, 2005: Drilling of the first new monitoring well was initiated (Monitoring Well-30).

13. January 13, 2006: A permanent leak collection box was installed encompassing the Unit 2 SFP crack.

14. January 31, 2006: A NRC Special Inspection team met on-site to review the Phase 1 monitoring well hydrology results.

15. February 8-10, 2006: A NRC Special Inspection team was on-site to evaluate the licensee's compliance with IE Bulletin 80-10 (radiological monitoring of on-site non-contaminated systems), 10 CFR 50.75(g) (on-site spill documentation for future decommissioning), and chemistry counting quality control requirements. Hudson River waterfront well sample splits were taken for NRC, NYS and IPEC.

16. February 27, 2006: Monitoring Well-37 initial sample result = 30,000 pCi/L, provided the first indication of a tritium groundwater release directly to the Hudson River.

17. February 28, 2006: Licensee provided a revised dose calculation of 0.000015 mrem/yr to the maximally exposed member of the public based on a general site area hydrology water transport and multiple contamination area drainage model. The NRC conducted the SIT exit meeting.

18. March 16, 2006: NRC Special Inspection Report No. 05000247/2005001 was issued describing NRC=s initial response and evaluation of the Indian Point groundwater contamination issue.

19. March 21, 2006: NRC sample results of Monitoring Well-37 strontium-90 analyses were received indicating 26 pCi/L. This was the first indication that strontium-90 was likely being released directly to the Hudson River. Initial bounding calculations were revised, indicating less than 0.1% of effluent release limits.

20. April 1, 2006: Due to the 2/21/06 discovery of strontium-90 in Monitoring Well-111, the licensee initiated continuous demineralization of the Unit 1 SFP in order to reduce the leaking source term.

21. April 10, 2006: Entergy groundwater monitoring and commitment letter sent to NRC Region I.

22. April 24, 2006: Updated dose assessment based on 2/28/2006 methodology using more recent monitoring well data and maximum concentrations of hydrogen-3 (tritium), strontium-90 and nickel-63: 0.0025 mrem total body and 0.011 mrem maximum organ (adult bone).

23. June 12-16, 2006: NRC groundwater contamination hydrology inspection team was on-site. U.S. Geological Survey participation was added to the NRC inspection effort.

24. November 7, 2006: NRC split sample results identify licensee strontium-90 results from 8/1 - 9/18/2006 were low and caused licensee resampling and licensee investigation.

25. October 30- November 1, 2006: Entergy conducted a 3-day groundwater draw-down pump test from Recovery Well - 1 (adjacent to Unit 2 SFP).

26. November 13-17, 2006: NRC on-site team inspection to review Unit 1 SFP leak history and

hydrology results of a 3-day pump down test of RW-1.

27. February 8, 2007: Fluorescein dye tracer test injected near the base of Unit 2 SFP. Test samples were collected through August 2007.

28. March 21, 2007: NRC inspection team reviewed preliminary tracer test results.

29. May 9-10, 2007: NRC conducted an on-site inspection team review of tracer test results and the evaluation of groundwater transport.

30. June 6-22, 2007: An expanded control zone fish split sampling exercise was conducted to include a second control location in the Catskills to help evaluate background levels of strontium-90 in fish.

31. June 2007: The Unit 2 SFP transfer canal was drained below the pinhole leak, which arrested this leak pathway.

32. July-August 2007: An independent fracture flow analysis using down hole geophysical and flow logs was conducted by the USGS to compare groundwater flow results based on fracture flow with the licensee=s groundwater flow rate calculations derived from packer testing data (slug tests) and based on a general porous media groundwater flow model.

33. August 31, 2007: The last monitoring well was installed and became operational (Monitoring Well-67).

34. November 7-9, 2007: NRC inspection team was on-site to compare and review the final site conceptual groundwater model based on all previously derived site data and USGS analyses.

35. December 15, 2007: The pinhole leak in the Unit 2 SFP transfer canal was repaired.

36. January 14, 2008: NRC received Entergy’s final site hydrogeological investigation report.

37. January 29, 2008: NRC received Entergy’s Synopsis of Long Term Monitoring Plan Bases.

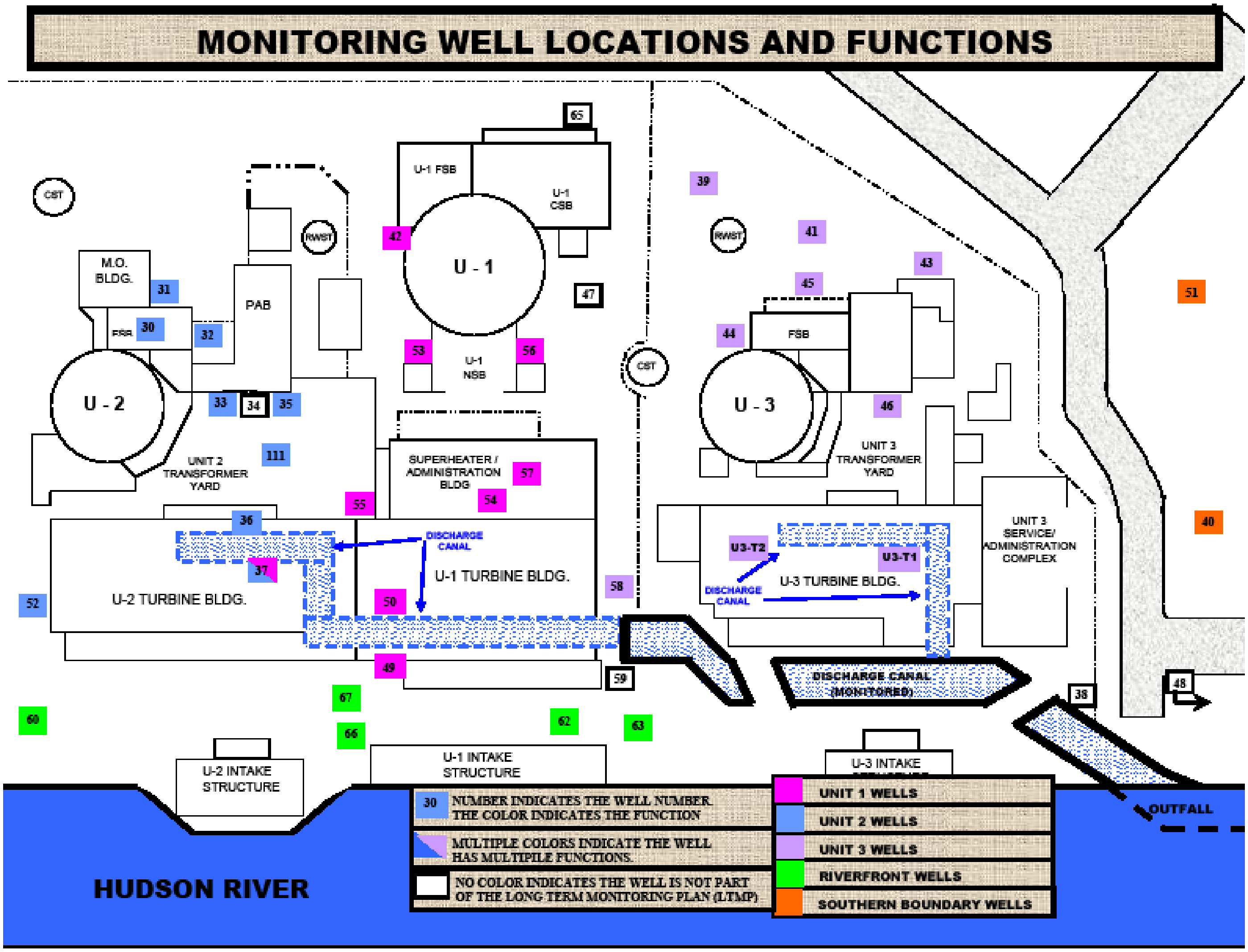
38. February 4, 2008: NRC inspection team conducted a critique of the Long Term Monitoring Plan and associated implementing procedures.

39. February 21, 2008: NRC held a meeting with Entergy and GZA to discuss further development and refinement of the Long Term Monitoring Plan.

40. May 7, 2008: NRC conducted an exit meeting of inspection report 50-003/2007010 & 50-247/2007010.

ATTACHMENT 2

Site Groundwater Contaminant Concentrations



Indian Point Monitoring Well Groundwater Contamination
Results as of 12/31/2007 in units of pCi/L

	H-3		Sr-90		Ni-63	Cs-137
Southern Boundary Wells						
MW-40	ND		ND		ND	ND
MW-51	ND		ND		ND	ND
Northern Boundary Wells						
MW-52	ND		ND		ND	ND
MW60	ND		ND		ND	ND
Eastern Boundary Well						
MW-65	ND		ND		ND	ND
Riverfront Wells						
MW-60	ND		ND		ND	ND
MW-66	9000		11		ND	ND
MW-67	5000		27		ND	ND
MW-62	780		2		ND	ND
MW-63	ND		ND		ND	ND
Unit 2 SFP Wells						
MW-30	130000		ND		ND	3000*
MW-31	36000		ND		ND	200*
MW-32	14000		ND		ND	ND
MW-33	23000		ND		ND	ND
MW-34	22000		ND		ND	ND
MW-35	6000		ND		ND	ND
MW-111	100000		1		ND	ND
MW-36	12000		2.5		ND	ND
MW-37	6000		28		56	ND
MW-55	10000		32		ND	ND
MW-50	4000		47		ND	ND
MW-49	7000		26		ND	ND
Unit 1 SFP Wells						
MW-42	2500		47		200	37000
MW-53	7400		28		ND	ND
MW-55	10000		32		ND	ND
MW-50	4000		47		ND	ND
MW-49	7000		26		ND	ND
MW-47	3500		4		ND	ND
MW-56	1500		2		ND	ND

Attachment

MW-57	4000		38		ND	ND
MW-54	2000		20		ND	ND
MW-58	900		ND		ND	ND
MW-59	800					
Unit 3 Wells						
MW-39	ND		5		ND	ND
MW-41	ND		6		ND	ND
MW-45	2200		ND		ND	ND
MW-44	ND		ND		ND	ND
MW-43	ND		ND		ND	ND
MW-46	1700		ND		ND	ND
U3-T1	530		ND		ND	ND
U3-T2	1200		ND		ND	ND
Off-site Locations						
LaFarge No. 1	ND		ND		ND	ND
LaFarge No. 2	ND		ND		ND	ND
LaFarge No. 3	ND		ND		ND	ND
Trap Rock Quarry	ND		ND		ND	ND
5th Street Well	ND		ND		ND	ND
Camp Field Reservoir	ND		ND		ND	ND
New Croton Reservoir	ND		ND		ND	ND
ND indicates nothing detectable above background						

* Single positive result was obtained immediately after a 3-day pump down test indicating hydraulic connectivity between Monitoring Well-42 and Monitoring Well-30 and 31.

These radionuclide concentrations reflect end of 2007 results. Due to annual cyclic groundwater flow variability, no definite trend of the radionuclide concentrations could be conclusively determined at the present time. Additional sample data over time will clarify whether the Unit 1 and Unit 2 groundwater plumes are shrinking in size or concentration.

ATTACHMENT 3

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

M. Barvenik	Principal Engineer, GZA Geo Environmental, Inc.
J. Comiotes	Director, Nuclear Safety Assurance
P. Conroy	Manager, Licensing
D. Croulet	Licensing Engineer
P. Donahue	Chemistry Specialist
J. Pollock	Site Vice President
C. English	Unit 1 Project Engineer
G. Hinrichs	Project Engineer
D. Loope	Radiation Protection Superintendent
T. Jones	Licensing Engineer
R. LaVera	Radiological Engineer
D. Mayer	Director, Special Projects
J. Peters	Plant Chemist
S. Sandike	Chemistry ODCM Specialist

New York State Inspection Observers

T. Rice	Environmental Radiation Specialist, New York State, Department of Environmental Conservations (NYS DEC)
L. Rosenmann	Engineering Geologist, NYS DEC
A. Czuhanich	Engineering Geologist, NYS DEC

LIST OF INSPECTIONS PERFORMED

7112203	Radiological Environmental Monitoring Program and Radioactive Material Control

LIST OF DOCUMENTS REVIEWED

Entergy Letter, NL-08-009 to USNRC, “Results of Ground Water Contamination Investigation,” January 11, 2008

GZA Final Report Hydrogeologic Site Investigation Indian Point Energy Center, January 7, 2008

GZA Memorandum to Entergy, “Synopsis of Long Term Monitoring Plan Bases,” January 25, 2008

Consolidated Edison Calculation No. CGX-00006-00, ASeismic Qualification Structural Evaluation of the Unit 2 Fuel Pool Wall Considering Deteriorated Condition of Concrete Due to Pool Leak@

United Engineers and Constructors Technical Report No. 8281,@Evaluation of Spent Fuel Pool Walls - Indian Point 2 Nuclear Power Plant@

ABS Consulting Report 1487203-R-001, AStudy of Potential Concrete Reinforcement Corrosion on the Structural integrity of the Spent Fuel Pit@September 2005

Chazen, ANorthern Westchester County groundwater conditions summary, data gaps and program recommendations,@Contract C-PL-02-71, Dutchess County Office, the Chazen Companies, Poughkeepsie, NY, April 2003

Clark, J.F., P. Schosser, M. Stute, and H.J. Simpson, ASF_6 - ^{3}He tracer release experiment: A new method of determining longitudinal dispersion coefficients in large rivers,@*Environmental Science and Technology*, vol 30, pp 1527-1532, 1996

Annual Radiological Environmental Operating Reports, 2005 and 2006

Radioactive Effluent Release Reports, 2005 and 2006

Pre-Operational Environmental Survey of Radioactivity in the vicinity of Indian Point Power Plant, 1958 and 1959

SECY-96-001, Order to Authorize Decommissioning and Amendment to License No. DPR-5 for Indian Point Unit No. 1, January 2, 1996

Indian Point Nuclear Generating Unit No. 1, License Amendment No. 42 and Technical Specifications

de Vries, P, and L.A. Weiss, ASalt-front movement in the Hudson River Estuary, New York - simulations by one-dimensional flow and solute-transport models,@U.S. Geological Survey, Water Resources Investigations Report 99-4024, 2001

Freeze and Cherry, *Groundwater*, 1979

GWPO, AGroundwater Program Office annual report for fiscal year 1994, ORNL/GWPO-013

NCRP, AScreening Models for Releases of Radionuclides to Atmosphere, Surface Water and Ground,@National Council on Radiation Protection and Measurements, Report No. 123, 1996

Whitman, AAssessment of groundwater migration pathways from Unit 1 spent fuel pools at Indian Point Nuclear Power Plant,@the Whitman Companies Inc, Project 940510, July 1994

ABS Consulting Report 1394669-R-004, Rev. C, AAssessment of Leakage from Unit 1 West Fuel Pool during Fuel Cleaning Activities@

ABS Consulting Report 1186959-R-007, April 2004,"Indian Point Unit 1 East Spent Fuel Pool and Rack Fitness for Service Inspection Report"

ENN-DC-114, Rev. 2, AUnit 1 Remediation - Phase 1 Project Plan"

USGS Open File Report 01-385, ACharacterization of Fractures and Flow Zones in a Contaminated Shale of the Watervliet Arsenal, Albany County, NY@

Procedures

EN-LI-102, ACorrective Action Process@Rev. 3
EN-LI-118, ARoot Cause Analysis Process@Rev. 3
EN-LI-119, AApparent Cause Evaluation (ACE) Process@Rev. 3
HP-SQ-3.013, Rev. 12, ARoutine Surveys Outside the Normal RCA@
2-CY-2625, Rev. 9, AGeneral Plant Systems Specifications and Frequencies@
3-CY-2325, Rev. 6, ARadioactive Sampling Schedule@
IPEC IE Bulletin 30-10 Program
O-CY-1510, Rev. 3, “IPEC Storm Drain Sampling”
O-CY-2740, Rev. 0, “Liquid Radiological Effluents”
O-CY-1420, Rev. 1, “Radiological Quality Assurance Program”
O-RP-NEM-101, Rev. 0, “Nuclear Environmental Monitoring Sampling and Analysis Schedule”
O-RP-NEM-100, Rev. 0, “Notification, Investigation and Reporting of Abnormal Activity in Environmental Samples”
IP-SMM-CY-110, Rev. 0, “Radiological Groundwater Monitoring Program”
GZA-IP-101, Rev. 0, “Radiological Groundwater Monitoring Program Quality Assurance and Procedures IPEC”
IPEC Off-site Dose Calculation Manual

Condition Reports

IP2-2005-03885
IP2-2005-03557
IP2-2005-04151
IP2-2005-03986
IP2-2005-04152
IP2-2005-M-11
IP2-2005-04789
IP2-2005-04799
IP2-2005-04957
IP2-2005-04977
IP2-2005-05145
IP2-2005-05160
IP2-2005-05194
IP2-2006-00137
IP2-2006-00488

Drawings
9321-F-1196-7, Fuel Storage Building Concrete Details No. 1
9321-F-1197-8, Fuel Storage Building Concrete Details No. 2
9321-F-1198-8, Fuel Storage Building Concrete Details No. 3
9321-F-1199-7, Fuel Storage Building Concrete Details No. 4
9321-F-1200-5, Fuel Storage Building Concrete Details No. 5

9321-F-1388-15, Fuel Storage Building Floor Plans, Section & Roof
9321-F-1389-11, Fuel Storage Building - Building Elevations & Section
9321-F-1390-05, Fuel Storage Building - Building Details & Door Schedule
9321-F-2514-16, Fuel Storage General Arrangement Plans & Elevations (U2)
9321-F-2576-24, Fuel Storage Building Auxiliary Coolant System Plans
9321-F-2577-24, Fuel Storage Building Auxiliary Coolant System Sections
9321-F-2715-5, Containment Building Piping & Penetrations - Details of Fuel Transfer Tube
9321-F-2762-15, Fuel Storage Building Piping Supports

Miscellaneous
ENN-LI-101 Att. 9.1, 50.59 Screen Control Form Activity, ID No. DCP-03-2-128
IP2 FSAR, Section 1.2.1.2, AGeology and Hydrology@ Rev. 19
IPEC Preliminary Cause Analysis, FSB Concrete Wall/Tritium in the Groundwater, February 10, 2006

NRC Groundwater Sample Result Documentation

ML060720148	ML061880387	ML062720227	ML070110577
ML070110602	ML070110559	ML070110548	ML070110561
ML070940618	ML070940504	ML070940574	ML070940515
ML070940546	ML070940534	ML071900442	ML071900462
ML071900438	ML071900445	ML071900447	ML071900458
ML072840255	ML071900448	ML071900456	ML072840312
ML072840323	ML072840334	ML072840357	ML072840292
ML072840278	ML080080499	ML073180148	ML073180167
ML073620089			

LIST OF ACRONYMS

CFR	Code of Federal Regulations
CR	condition report
CSM	conceptual site model
DEC	State of New York Department of Environmental Conservation
EDO	Executive Director for Operations
EPA	Environmental Protection Agency
ESSAP	Environmental Site Survey and Assessment Program
FSAR	final safety analysis report
FSB	Fuel Storage Building
GPD	gallons per day
GPM	gallons per minute
IN	Information Notice
IP	Inspection Procedure
IP2	Indian Point 2
IPEC	Indian Point Energy Center
IR	Inspection Report
ISFSI	independent spent fuel storage installation
MDC	minimum detectable concentration
MSL	mean sea level
MW	monitoring well
NCD	north curtain drain
NYS DEC	State of New York Department of Environmental Conservation
NYSEMO	State of New York Emergency Management Organization
NYSPSC	State of New York Public Services Commission
ORISE	Oak Ridge Institute for Science and Education
PCB	polychlorinated biphenyls
pCi/L	pico-Curies per Liter
REMP	Radiological Environmental Monitoring Program
SFD	sphere foundation drain
SFP	spent fuel pool
USGS	United States Geological Survey

Note: Explanation of the terms groundwater, ground-water and ground water -- Hydrologists often use the term Aground-water@ in adjective form and Aground water@ in noun form. This report has not followed that convention, and instead typically uses Agroundwater@ universally. However, all three forms of the word may be used herein.

May 9, 2008

Vice President, Operations
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, NY 10511-0249

SUBJECT: INDIAN POINT ENERGY CENTER, UNIT 1 - ISSUANCE OF AMENDMENT REGARDING USE OF A NON-SINGLE FAILURE PROOF CRANE FOR SPENT FUEL CASK HANDLING OPERATIONS, FINAL SAFETY ANALYSIS REPORT SECTIONS 3.5.1 AND 6.0 (TAC NO. J00293)

Dear Sir or Madam:

The Commission has issued the enclosed Amendment No. 53 to Facility Operating License No. DPR-5 for the Indian Point Energy Center, Unit 1. The amendment consists of changes to the Final Safety Analysis Report (FSAR) in response to your application dated February 27, 2007, as supplemented by letters dated October 3, 2007 and February 27, 2008.

The amendment enables the licensee to make changes to the FSAR to reflect the use of the non-single-failure-proof Fuel Handling Building crane main hoist for dry storage cask component lifting and handling operations, in support of transfer of spent fuel to dry storage in an Independent Spent Fuel Storage Installation.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Theodore B. Smith, Project Manager
Reactor Decommissioning Branch
Decommissioning and Uranium Recovery
  Licensing Directorate
Division of Waste Management
  and Environmental Protection
Office of Federal and State Materials and
  Environmental Management Programs

Docket No.: 50-003

Enclosures:
1. Amendment No. 53 to DPR-5
2. Safety Evaluation

cc: Indian Point-1 Service List w/o Enclosure 2

May 9, 2008

Vice President, Operations
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
450 Broadway, GSB
P. O. Box 249
Buchanan, NY 10511-0249

SUBJECT: INDIAN POINT ENERGY CENTER, UNIT 1 - ISSUANCE OF AMENDMENT REGARDING USE OF A NON-SINGLE FAILURE PROOF CRANE FOR SPENT FUEL CASK HANDLING OPERATIONS, FINAL SAFETY ANALYSIS REPORT SECTIONS 3.5.1 AND 6.0 (TAC NO. J00293)

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/RA/

Theodore B. Smith, Project Manager
Reactor Decommissioning Branch
Decommissioning and Uranium Recovery
Licensing Directorate
Division of Waste Management
and Environmental Protection
Office of Federal and State Materials and
Environmental Management Programs

Docket No. 50-003

Enclosures:
1. Amendment No. 53 to DPR-5
2. Safety Evaluation

cc: Indian Point-1 Service List w/o Enclosure 2

DISTRIBUTION:
RDB r/f RLorsen, RI LKauffman, RI

ML081080243

OFFICE	DWMEP:PM	DWMEP:LA	DWMEP:SC	SFTS	OGC	DWMEP:DD	DWMEP
NAME	TSmith	TMixon	APersinko	GBjorkman	JHull	KMcConnell	TSmith
DATE	4/2/08	4/3/08	5/8/08	4/25/08	4/22/08	5/9/08	5/9/08

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ENTERGY NUCLEAR OPERATIONS

DOCKET NO. 50-003

INDIAN POINT GENERATING UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 53
License No. DPR-5

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Nuclear Operations, Inc., (Entergy, the licensee), dated February 27, 2007, as supplemented October 3, 2007, and February 27, 2008, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will be maintained in conformity with the application, as amended, the provisions of the Act, and the applicable rules and regulations of the Commission;

   C. There is reasonable assurance: 1) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public; and 2) that such activities will be conducted in compliance with applicable portions of the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, by Amendment No. 53, the license is amended to authorize revision to the Final Safety Analysis Report (FSAR), as set forth in the application for amendment by Entergy dated February 27, 2007, as supplemented by letters dated October 3, 2007, and February 27, 2008. Entergy shall update the FSAR by the next periodic update, to reflect the revisions authorized by this amendment in accordance with 10 CFR 50.71(e).

3. The license amendment is effective as of its date of issuance, with the implementation to begin immediately and completed by the next periodic update to the FSAR in accordance with 10 CFR 50.71(e). Implementation of the amendment is the incorporation into the FSAR the revisions described in the amendment application of February 27, 2007, as supplemented by letters dated October 3, 2007, and February 27, 2008, and evaluated in the NRC staff's Safety Evaluation attached to this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Keith I. McConnell, Deputy Director
Decommissioning and Uranium Recovery
Licensing Directorate
Division of Waste Management
and Environmental Protection
Office of Federal and State Materials and
Environmental Management Programs

Date of Issuance: May 9, 2008

SAFETY EVALUATION BY OFFICE OF FEDERAL AND STATE MATERIALS

AND ENVIRONMENTAL MANAGEMENT PROGRAMS

RELATED TO AMENDMENT NO. 53 TO FACILITY OPERATING LICENSE NO. DPR-5

ENTERGY NUCLEAR OPERATIONS, INC.

INDIAN POINT ENERGY CENTER, UNIT 1

DOCKET NO. 50-003

1.0 INTRODUCTION

By application dated February 27, 2007, (Agencywide Document and Access Management System (ADAMS) Accession No. ML070740552), as supplemented by letters dated October 3, 2007, (ADAMS Accession No. ML073050247), and February 27, 2008, (ADAMS Accession No. ML080630507), Entergy Nuclear Operations, Inc. (Entergy, the licensee) requested changes to the Final Safety Analysis Report (FSAR) for the Indian Point Energy Center, Unit 1 (IP-1). The supplements dated October 3, and February 27, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on June 19, 2007 (72 FR 33779).

The amendment would enable the licensee to make changes to the FSAR to reflect the use of the non-single-failure-proof Fuel Handling Building (FHB) crane main hoist for dry spent fuel cask component lifting and handling operations. Specifically, the FHB crane main hoist will be used for the lifting and handling of the spent fuel canister, canister lid, and transfer cask, as needed. This amendment would change the IP-1 FSAR to reflect this proposed use of the FHB crane main hoist. The licensee has stated that the FSAR will be revised to include a summary of the activities in support of dry spent fuel storage that take place in the IP-1 FHB, and to add a discussion related to the spent fuel storage cask component drops.

2.0 BACKGROUND

IP-1 was permanently shut down in October 1974, and is currently in safe storage condition (SAFSTOR). SAFSTOR is the decommissioning method in which a nuclear facility is placed and maintained in a condition that allows the safe storage of radioactive components of the nuclear plant and subsequent decontamination to levels that permit license termination. A Decommissioning Plan (DP) was approved in January 1996. Subsequent to the 1997 decommissioning rule, the licensee converted its DP into its FSAR.

NRC Bulletin 96-02, “Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor, or Over Safety-Related Equipment” requested licensees to review and report their plans and capabilities for handling heavy loads in accordance with existing regulatory guidelines and within their licensing basis as previously analyzed in their FSAR. In response, IP-1 committed to submit a License Amendment Request (LAR) should spent fuel cask handling operations be resumed in the IP-1 Fuel Handling Building, and if a cask was to be lifted over the spent fuel pool or in a manner otherwise outside the licensing basis.

The licensee, in its letter dated February 27, 2007, provided the background as summarized in the following subsections.

2.1 Fuel Handling Building 75-ton Crane Design and Licensing Considerations

The IP-1 FHB 75-ton crane was designed and procured in 1958 and installed in the FHB for plant startup in 1962. It is a non-safety-related, commercial-grade crane originally designed and licensed to lift and handle a spent fuel shipping cask. The crane was used to load and transport 124 Core A fuel assemblies and 120 Core B fuel assemblies from IP-1 to an off-site fuel reprocessing facility. The loading and transport, which utilized two casks and was completed without incident, required several hundred lifts of the 30-ton shipping casks. The crane was also used to lift and move individual spent fuel assemblies within the spent fuel pool and between the spent fuel pool and the cask loading pool.

The FHB 75-ton crane is a bridge-and-trolley design that is not single-failure-proof as defined in NUREG-0612, "Control of Heavy Loads at Power Plants," Resolution of Generic Technical Activity A-36, issued July 1980 (NUREG-0612) and its design pre-dates publication of that document. However, many of the criteria contained in NUREG-0612 pertaining to crane design, maintenance, and inspection, as well as operator training, safe load paths, and design of lifting devices associated with cask handling are, or will be met, as part of the dry cask storage project at IP-1. The FHB 75-ton crane main hoist has a rated load of 75 tons and the auxiliary hoists have a rated load of 15 tons and 3 tons respectively. Only the main hoist is used to lift the transfer casks.

A review of the crane design, maintenance, and operational history was performed. This review concluded that with additional analysis, inspections, and load testing, the 75-ton HI-TRAC 100D-Version IP1™ transfer cask loaded with spent fuel can be handled by the 75-ton crane main hoist with a load drop being a highly unlikely event. An analysis was performed to demonstrate the crane can handle the rated load under the limiting loading conditions including a seismic event concurrent with a loaded cask suspended from the main hook. Inspections of welds, bolting, and structural steel were performed to provide reasonable assurance that the crane was installed according to the design drawings and specifications. However, consistent with the guidance in NUREG-0612, because the crane is not single failure proof, several hypothetical drops of heavy loads associated with cask loading have been analyzed to ensure their consequences were acceptable.

The HI-TRAC 100D-Version IP1™ transfer cask and the Multiple Purpose Canister (MPC) must be lifted and moved several times during fuel loading operations in the FHB. At various points in the operation, the empty transfer cask with the empty MPC, the MPC lid, and the fuel-loaded transfer cask and MPC must be lifted and handled by the FHB 75-ton crane. Five lifts out of the cask load pool of a fuel-loaded transfer cask and MPC, approaching the 75-ton design rating of the crane, are planned to be performed to place the transfer cask and MPC at the cask preparation work station.

Five additional lifts of the fuel-loaded transfer cask and seal welded MPC are planned to place the transfer cask and sealed MPC on the air transporter pad for movement out of the FHB. These five additional lifts are a few inches in height, over the FHB truck bay floor.

The locations where drops are postulated and evaluated comply with the applicable Part 50 licensing requirements, and are consistent with the guidance in NUREG-0612, NRC Bulletin 96-02, and Regulatory Issues Summary RIS-2005-25.

To mitigate the consequences of two of the postulated transfer cask drops, namely, the vertical cask drop into the cask load pool and a tilted transfer cask drop also into the cask load pool, an impact limiter will be employed on the floor of the cask load pool to limit the g forces to which the transfer cask, MPC, and fuel in the loaded MPC would be exposed. The impact limiter serves no Part 50 design function since (1) the transfer cask and MPC will not be lifted over or near (within 15 ft) the spent fuel pool, (2) there is no safety related or essential-to-operation equipment in the FHB, and (3) the entire travel path of the load is exclusively over concrete floors founded directly on bed rock or engineered fill.

FHB Loading Operations Summary

The HI-STORM 100S Version B, Type 185 System™ will be used for dry cask storage of IP-1 spent fuel at the IPEC Independent Spent Fuel Storage Installation (ISFSI). This IP-1 custom-designed dry spent fuel storage system is currently under NRC review as a proposed amendment to the Holtec HI-STORM 100 System 10 CFR 72 Certificate of Compliance (CoC). The ISFSI will serve as the temporary storage facility for spent fuel from IPEC Units 1, 2, and 3 until such time as the fuel is removed from the site and sent to a federal repository. Five of the storage systems will contain all of the fuel from the “SAFESTOR” Unit 1, and the remaining storage systems will contain fuel from operating Units 2 and 3.

The HI-STORM 100S Version B, Type 185 System™ consists of a multi-purpose canister (MPC-32) which is capable of holding 32 IP-1 fuel assemblies; a Transfer Cask (HI-TRAC 100D-Version IP1™), which contains the MPC during loading, unloading, and transfer operations; and a storage overpack (HI-STORM 100S Version B, Type-185™), which provides shielding, heat removal capability, and structural protection for the MPC during storage operations at the ISFSI. IP-1 fuel is stainless steel clad and the fuel rods are shrouded in a protective sheath. The assemblies are nominally 137 in. long. Because of the unique configuration of the IP-1 fuel, the five HI-STORM 100S, Version B, Type185™ systems are specifically designed for IP-1 and are not interchangeable with Units 2 and 3 or any other pressurized water reactor (PWR).

The 75-ton crane is required to lift and handle the HI-TRAC transfer cask and MPC (both empty and loaded with spent fuel) and the MPC lid in support of dry cask loading. The combined maximum lift weight, including rigging and lift yoke will not exceed 75 tons, which is the design rated load of the IP-1 FHB 75-ton crane.

All the IP-1 fuel is stored in the west fuel pool. The east fuel pool is currently not used for fuel storage. IP-1 also has disassembly and cask loading pools.

During each of the five cask loading evolutions, spent fuel assemblies are moved one at a time, from the spent fuel racks in the west pool, through a gate slot into the disassembly pool, through a second gate slot into the cask load pool and into the MPC. The cask load pool and the disassembly pool will have been previously flooded to the same level as the west spent fuel pool and the intermediate gates will have been removed. Once the MPC is loaded with 32 fuel assemblies, the MPC lid is installed under water, and the transfer cask with the loaded MPC inside is lifted by the 75-ton crane and placed on the 70 ft 6 in. floor to the east of the cask load pool. The horizontal cask trolley movement totals approximately 22 ft from the cask load pool position to the cask preparation work station in the adjacent area to the east.

At the 70 ft 6 in. location, the MPC is seal welded and the canister is drained, dried, and backfilled with helium and leak tested in accordance with the 10 CFR 72 Certificate of Compliance (CoC) and cask FSAR. The transfer cask containing the sealed MPC is lifted a few inches and placed on an air castor and moved northward out of the FHB. A Vertical Cask Transporter (VCT) will move the transfer cask to the Unit 2 FHB where it is placed on top of an empty storage overpack using the Unit 2 single-failure-proof gantry crane. The Unit 2 gantry crane is disengaged from the transfer cask lifting trunions and rigged to lift the MPC, which is inside the transfer cask. The MPC is lifted to take its weight off the transfer cask pool lid, which is then removed, and the MPC is lowered through the transfer cask into the overpack. After the overpack lid is installed, the HI-STORM overpack is transported to the ISFSI using the VCT.

The evolutions performed in the Unit 2 FHB with the gantry crane are essentially identical to the operations associated with the Unit 2 fuel handling operations reviewed and approved by the NRC on November 21, 2006 (ADAMS Accession No. ML053000051), except for the IP-1 MPC and transfer cask being shorter and lighter than the corresponding Unit 2 components.

3.0 REGULATORY EVALUATION

The regulatory requirements and regulatory guidance on which the U.S. Nuclear Regulatory Commission (NRC) staff based its review are discussed below:

NUREG-0612 provides guidelines and recommendations to assure safe handling of heavy loads by prohibiting, to the extent practicable, heavy load travel over stored spent fuel assemblies, fuel in reactor core, safety-related equipment, and equipment needed for decay heat removal.

NUREG-0612 provides the basis for review of the licensee-proposed handling of heavy loads during the dry spent fuel cask loading operation. NUREG-0612 endorses a “defense-in-depth” approach for handling of heavy loads near spent fuel and safe shutdown systems to minimize load handling accidents and their consequences. General guidelines for overhead handling systems that are used to handle heavy loads in the area of the reactor vessel and spent fuel pool are given in Section 5.1.1 of NUREG-0612. They are as follows: (1) definition of safe load paths; (2) development of procedures for load handling operations; (3) training and qualification of crane operators in accordance with Chapter 2-3 of American National Standards Institute (ANSI) B30.2-1976; (4) use of special lifting devices that meet guidelines in ANSI N14.6-1978; (5) installation and use of non-custom lifting devices in accordance with ANSI B30.9-1971; (6) inspection, testing, and maintenance of cranes in accordance with Chapter 2-2 of ANSI B30.2-1976; and (7) design of crane in accordance with Chapter 2-1 of ANSI B30.2-1976 and Cranes Manufacture of America (CMMA) Standard 70 (CMMA-70), "Overhead and Gantry Cranes (1967)."

NUREG-1864, “*A Pilot Probabilistic Risk Assessment Of a Dry Cask Storage System At a Nuclear Power Plant*”, Issued March 2007 (NUREG-1864), develops and assesses a comprehensive list of initiating events, including dropping the cask during handling and external events during onsite storage (such as earthquakes, floods, high winds, lightning strikes, accidental aircraft crashes, and pipeline explosions), and models potential cask failures from mechanical and thermal loads.

4.0 TECHNICAL EVALUATION

The technical evaluations of the proposed amendment considered welding and nondestructive examinations of the crane, structural adequacy of the crane, load drop analysis of the spent fuel cask and components, and load drop analyses of the spent fuel.

4.1 Welding and Nondestructive Examination of the Crane

Materials and Fabrication

Section 4.3 (Crane Structural Steel) of Attachment 1 to the licensee's license amendment request dated February 27, 2007 (ML070740552), stated that an engineering review of the crane's past inspection and maintenance history was performed. As a result of this review and in preparation for the planned dry cask loading effort, critical structural areas were identified and inspected. In its supplemental letter dated October 3, 2007 (ML073050247), the licensee stated that the critical components were identified based on (1) analysis results of the integrated crane and building model showing locations of high loading (e.g., bridge girder bolting to end trucks); (2) locations of significance from a primary load path standpoint (e.g., main hook); (3) locations of significance from a basic structural mechanics/strength of materials standpoint (e.g., bridge girder welds at mid-span cross-section extremes); and (4) locations of significance based on structural engineering experience (e.g., end truck bolting securing wheel bearing housing). The nondestructive examinations (NDE) comprised of performing visual testing (VT), ultrasonic testing (UT) and magnetic particle testing (MT) at the identified inspection locations. The inspections were performed using Entergy procedures by Level 2 inspectors. This inspection confirmed that all of the selected critical welds were in acceptable condition. However, the inspection identified cracking in some bridge rail tie-down bolting which were made of copper alloy 655 material (currently alloy UNS (Unified Numbering System) C65500). All such bolting was replaced with ASTM (American Society for Testing and Materials) A-307 bolting. The staff finds that the results of this review and inspection will provide reasonable assurance that the crane structural steel components will perform as designed.

Section 4.3 also stated that, as a result of the seismic qualification effort, the trolley-to-end truck bolting was replaced to assure the adequacy of the bolting to resist the calculated seismic stresses. In its supplemental letter dated October 3, 2007, the licensee stated that the reason for the replacement is due to the finding that the crane drawings do not identify the bolt material. Based on a search of available materials during the period of crane assembly, it was determined that the bolts could be made by either ASTM A-7 or ASTM A-325 material. ASTM A-7 material was found to not satisfy the design requirements of supporting the maximum load lift during a seismic event. Since the available NDE is not capable of confirming the material of the bolts, the licensee replaced all such bolts with ASTM A-325 bolts.

Section 4.4 (Crane Inspections and Tests) stated that the IP-1 FHB 75-ton crane received pre-use inspections, quarterly exercise and operational inspections and an annual inspection by Whitey Services based upon 29 CFR 1910, Section 179, "Overhead and Gantry Cranes" and the original manufacturer's specifications. In its supplemental letter dated October 3, 2007, the licensee stated that the most recent inspections were performed prior to and following the proof test. Whiting Services, Inc. performed the pre- and post proof load test inspection of the crane on March 2, 2007 and March 14, 2007, respectively. The proof test was performed on March 6,

2007. The lift load test block used was 125.6% of the 150,000 lbs (75 tons) rated crane load. NDE comprising of VT, MT or UT was performed to verify the condition of the critical structural components. The licensee stated that the following critical structural components were inspected by NDE after the proof test: main hook, east/west end girder and truck welds, east/west end trolley and girder bolting, east/west end truck and locator bolting, east/west end truck tie welds, bridge girder welds and end truck wheel bolting. All of the post-proof test inspections were acceptable. NRC staff finds that the results of the proof test and the post-proof test inspections will provide reasonable assurance that the structural integrity of the critical crane components will be maintained when lifting the load as designed.

In its supplemental letter dated October 3, 2007, the licensee stated that the original crane specification (MP-5830, dated July 11, 1958) required fabrication welding to be in conformance with the latest applicable ASTM specifications. No significant structural modifications have been made on the crane since original fabrication. However, weld records or information beyond the original purchase specification requirements could not be found. The licensee also stated that any minor post-installation welding on the crane structure has been done in accordance with Indian point site procedures referencing AWS (American Welding Society) Welding Code D1.1 (Structural Welding code-Steel). The NRC staff finds that the fabricated welds at the subject crane are acceptable because they are fabricated in accordance with the applicable ASTM specifications or AWS Welding Code D1.1 and that the quality of the fabricated welds was successfully demonstrated by the results of NDE inspection and the 125% proof test.

Based upon its review of the information submitted by the licensee, the NRC staff finds that the critical structural components associated with the IP1 FHB 75-ton crane are of acceptable quality and thus reasonable assurance of adequate structural integrity and safety exists to ensure the subject crane will perform as designed.

4.2 Structural Adequacy of the Crane

4.2.1 Background of the IP-1 FHB 75-ton Crane and Proposed Cask Handling Operations

The licensee provided the background of the FHB crane in Section 3.1 of the LAR. The licensee stated that the IP-1 FHB 75-ton crane was designed and procured in 1958 (without seismic considerations) and installed inside the non-safety-related FHB for plant startup in 1962. The crane is a commercial-grade bridge-and-trolley design that is non-safety-related and non-single-failure-proof as defined in NUREG-0612. The crane was originally designed, licensed and used to lift and handle the 30-ton GE IF-200 spent fuel shipping casks.

The licensee stated that the crane design and procurement process predated the issuance of 10 CFR 50 Appendix B, NUREG-0612, and associated Generic Letters and Regulatory Guides. As part of the IP-1 Dry Cask Storage project, a comprehensive evaluation was undertaken to review the FHB crane original design, maintenance and operational history details and compare them to what is recommended by NUREG-0612 and current standard practice, design, and operational guidance. From this review and the results of additional calculations (including seismic qualification), inspections, and testing, the licensee concluded that the crane and superstructure are suitable for use in IP-1 dry storage cask loading operations.

The licensee stated that many of the criteria contained in NUREG-0612 pertaining to crane design, maintenance, and inspection, as well as operator training, safe load paths, and design of lifting devices associated with cask handling are, or will be met, as part of the dry cask storage project at IP-1. The licensee provided a matrix comparing the IP-1 cask handling operation to NUREG-0612 criteria in Table 4 of the LAR. In Attachment Three of the LAR, the licensee made commitments that included implementation of the General Guidelines of NUREG-0612 with regard to the FHB 75-ton crane.

By letter of September 7, 2007 (ML072480497), NRC staff requested the licensee to provide information on the preventive measures/controls that it would have in place to ensure that the auxiliary hoists (rated loads of 15 tons and 3 tons) are not used inadvertently in the transfer cask handling operations. In its response dated October 3, 2007 (ML073050247), the licensee stated that the Unit 1 cask loading operations will be controlled by specific cask loading procedures which will clearly require the use of the main hoist for the transfer cask and cask lid lifts. The licensee's response also indicated that the Holtec designed and fabricated special lifting device has been specifically designed to mate with the main hoist sister hook. The subject lifting device is not compatible with the smaller auxiliary hoists.

The licensee's response to NRC staff questions indicates that the procedures and the mating design of the special lifting device for the cask lifts ensure that only the main hoist can be used for the cask lifts. The staff finds the response acceptable since it adequately addressed the staff's concern.

In Table 4 of Attachment 1 of the LAR, the licensee provided an evaluation of the FHB crane design in comparison to design requirements in Section 5.1.1(7). The licensee stated that the IP-1 FHB bridge crane was fabricated in 1958 in accordance with specification requirements that required the crane to be designed in accordance with safety standard provisions of New York State or any other codes applying to this type of equipment that meet the intent of the New York codes. Codes in effect at that time included American National Standards Institute (ANSI) B30.2-1943 and Crane Manufacturers Association of America (CMAA) standards (in effect in 1955). The crane has since been evaluated to the requirements of (American Society of Mechanical Engineers) ASME NOG-1, which utilizes similar design requirements to that of Chapter 2-2 of ANSI B30.2 and CMAA-70. Supplemental testing and inspections have been or will be performed to offset the material design requirements of ANSI B30.2 and CMAA-70, including a full load proof test at minimum operating temperature and NDE inspection of critical welds post proof test.

The licensee provided a summary description of the fuel building loading operations and cask handling sequence in Section 3.2 and Table 3, respectively, of the LAR. The load path in the FHB for the 75-ton crane is illustrated in Figures 1, 2 and 3 of the LAR. The 75-ton FHB crane is required to lift and handle the HI-TRAC transfer cask and MPC (both empty and loaded with spent fuel) and the MPC lid in support of dry cask loading. The combined maximum lift weight, including rigging and lift yoke will not exceed 75 tons, which is the design rated load of the IP-1 FHB 75-ton Crane.

Since the cask lift load is expected to approach the rated load of the crane, NRC staff requested additional information, by letter dated September 7, 2007 (ML072480497), with regard to the actual lift weight expected to be lifted for the proposed cask handling operations at IP-1.

The licensee responded to NRC staff by letter dated October 3, 2007. The licensee confirmed that the manufacturer's 75-ton rated load for the FHB crane is exclusive of the 7,294 pound weight of the lower hook block assembly. The licensee provided a breakdown (HI-TRAC transfer cask, MPC canister, fuel, and lifting yoke) of the maximum expected calculated lift load for the IP-1 cask handling operations for two controlling lifts: (i) the lift out of the cask pool, and (ii) lift onto the air pad outside the pool. The lift out of the cask pool governed with a maximum lift weight of 67.2 tons.

The licensee stated that these lift weights were calculated bounding weight estimates. The calculated weight provides over 10 percent margin to the rated load. The licensee noted that for the heaviest item (the transfer cask body), the actual delivered weight (58,240 lbs) was less than the calculated weight (62,636 lbs) by over 2 tons and using this actual weight would increase the margin to the rated load to over 13 percent.

The licensee further emphasized that an upper bound lift weight of 75 tons (equal to the rated load) was used in the structural and seismic analyses of the FHB crane. Also, a 125 percent proof test was successfully completed, in March 2007, on the FHB crane using stamped test weights totaling 188,495 lbs (94.25 tons). Since the proof test was conducted using actual known test weights (as opposed to calculated), it indicates that the 75-ton rated load of the crane has a margin of safety of at least 25 percent.

The staff finds that the margins listed above to the rated load are conservative and adequate to accommodate reasonable variations in actual load from the calculated maximum lift load. Further, the available safety margin (based on the 125 percent proof-test and considering an upper bound lift weight equal to the rated load) provides reasonable assurance that the crane capacity is adequate to perform the required lifts. Therefore, the licensee’s response is acceptable.

4.2.2 <u>FHB 75-ton Crane Refurbishment, Inspections and Tests</u>

In Section 4.2 of the LAR, the licensee stated that based upon a thorough review and inspection of the existing electrical system and controls, it decided to undertake an electrical refurbishment of the crane in early 2007. This refurbishment was performed and functionally tested, included replacement of the motors and related controls and installation of a new pendant with auxiliary switches for start/stop and bridge zone control. Operation in the “Safe” mode will now restrict bridge travel to the area north of the west spent fuel pool, thus assuring that the bridge cannot travel near the spent fuel pool during dry cask handling operations.

In Section 4.3 “Crane Structural Steel” of the LAR, the licensee stated that it completed an engineering review of the cranes past inspection and maintenance history in preparation for the planned dry cask loading effort. Critical structural areas were identified and inspected using Entergy NDE procedures and the acceptance criteria of the applicable ASME, ANSI, and AWS codes and standards, referenced in ASME NOG-1, “Rules for Construction of Overhead Cranes.” The NDE inspections (VT, UT and MT) confirmed that all of the selected critical welds were in acceptable condition. The inspection of certain bridge rail tie-down bolted connections raised concern with potential bolt cracking. All the subject bolting has been replaced. As a result of the seismic qualification effort (discussed further below), the trolley-to-end truck bolting was replaced to assure the adequacy of the bolting material to resist the calculated seismic stresses.

By letter dated September 7, 2007 (ML072480497), NRC staff requested additional information regarding inspections of the crane structural steel. In particular, NRC staff asked how age-related degradation effects were considered and evaluated for the crane structural steel. NRC staff requested a list of the welds that were selected as critical and subjected to NDE inspection, and specific confirmation of following as critical welds: (i) the welds of the truck structure that supports and aligns the crane bridge and trolley wheels on their respective runway rails; and (ii) the welds that align the wheel trucks relative to the bridge girders and (iii) welds in the bridge girders and trolley load girder.

In its above response to the NRC staff request, the licensee stated in an October 22, 2007 letter (ML073050247), that the crane has always been in a controlled indoor environment and subjected to loading conditions that rarely exceeded 40 percent of the rated load, which minimized any age-related degradation and fatigue effects. The response also indicates that the licensee performed NDE inspections on all critical welds and bolting on the accessible crane structural steel using a qualified inspector and approved site procedures. These inspections of welds, bolting, and structural steel yielded acceptable results and provide reasonable assurance that the crane was installed according to the design drawings and specifications. The NRC staff finds that the response fully addressed the staff's concerns and is therefore acceptable.

By letter dated September 7, 2007 (ML072480497), NRC staff requested additional information with regard to the Quality Assurance (QA) program that was used in the electrical refurbishment and bolt replacement activities.

The licensee's October 22, 2007 (ML073050247), response to NRC staff indicates that the electrical refurbishment and bolt replacement efforts on the non-safety-related FHB crane were performed under controlled site programs that provide a reasonable assurance of quality. Therefore, staff finds the response acceptable.

The licensee described the "Crane Inspections and Tests" in Section 4.4 of the LAR. The licensee stated that the IP-1 FHB 75-ton crane receives pre-use inspections, quarterly exercise and operational inspections and an annual inspection by Whitey Services based upon 29 CFR 1910, Section 179, "Overhead and Gantry Cranes" and the original manufacturer's specifications. The use of the 75-ton crane to move and prepare low level radioactive waste for shipment to off site disposal facilities drives the inspection and maintenance requirements. An annual inspection was completed in August 2006 as part of the effort to evaluate and document the condition of the crane. The ongoing inspections and the 2007 electrical refurbishments and testing will ensure the ability of the crane to safely carry its 75 ton critical load.

By letter dated September 7, 2007 (ML072480497), NRC staff requested additional information with regard to the full load proof test of the crane. Staff requested the licensee to discuss the procedure and/or standard that will be used for performing the full load proof test of the crane, since the FHB 75-ton crane is an older vintage, (designed and procured in 1958 and installed in 1962) partly refurbished crane, and the limiting dry spent fuel cask load that will be handled is at or close to the design rated load of 75 tons. Additionally, NRC staff requested the licensee to discuss the basis for concluding that use of 100% of the design rated load of 75 tons as the proof test load for the crane provides a proper verification of the structural adequacy of the crane for dry spent fuel cask handling operations.

In response to NRC staff by letter dated October 3, 2007 (ML073050247), the licensee clarified that the actual proof load test was successfully conducted in March 6, 2007, at a load equal to 125.6% of the rated load, and not at 100% of the expected lift load. The test load consisted of 188,495 lbs of stamped (known) steel test weights. The higher 125% proof testing exceeds that required by more recent editions of ANSI B30.2 and is in agreement with the testing specified in NUREG-0612. The trolley was in a position that mimicked the position when lifting the dry casks from the cask load pool. The test load was lifted and held in position for five minutes with no indication of drift or distress. The temperatures of the crane steel and columns recorded during the test will be the lower bound temperatures permitted during the cask lifts. The test was conducted in a manner that is consistent with industry standards and meets the intent of NUREG-0612. Therefore, the staff finds that the licensee's response is acceptable.

4.2.3 <u>FHB 75-ton Crane Seismic Qualification</u>

The licensee's LAR states that, as part of the Dry Cask Storage Project's assessment of the crane, an analysis was performed which confirmed that the crane system and its supporting structure are qualified to hold the maximum critical load during a seismic Safe Shutdown Earthquake (SSE) event. Since there exist no specific IP-1 design seismic response spectra, the analysis utilized the 0.15 g ground response spectra specified for the adjacent Unit 2 and is consistent with previous seismic analyses performed for various IP-1 structures.

By letter dated September 7, 2007 (ML072480497), NRC staff requested the following additional information with regard to the determination of seismic response in the crane seismic qualification:

- Describe the methodology used for seismic qualification of the crane including the use of computer codes and models, and the limiting loads considered.

- Define the boundary of the crane system, and any crane configuration assumptions considered in the analysis and provide an explanation why the crane load has no impact outside of this boundary.

- Discuss the response spectra used, and its appropriateness as input for the crane seismic evaluation

- Indicate the approach used (time domain, frequency domain) for applying the seismic load to the crane structural model.

- Explain the treatment of the load on the hook in the seismic analysis for both horizontal and vertical seismic excitation effects.

- Explain how were seismically induced pendulum and swinging effects of the load considered in the analysis and design evaluation of the crane.

- Provide justification for any seismic effects not considered.

In an October 3, 2007 (ML073050247), response to NRC staff questions, the licensee stated that the seismic analysis of the crane system was performed by the response spectra method using the computer program SAP2000 Version 7.4. The licensee provided an appropriate rationale for use of the Unit 2 SSE response spectra for the analysis. The licensee provided a comprehensive description of the extent of the computer model, boundary conditions, components included, the governing trolley/hook positions, damping and the loads and load combinations considered in the seismic evaluation. The licensee also appropriately explained the rationale for considering only the vertical participation of the load on the hook to the seismic response. The licensee also justified that predicted response from the higher than actual stiffness modeled for the cable system is conservative. The staff finds that the approach and model used by the licensee for determining and evaluating the seismic response of the crane is reasonable, adequate and conservative. Therefore, the licensee's response is acceptable.

The licensee further stated, in Section 4.5 of the LAR, that the crane was evaluated in accordance with guidelines in NUREG-0612 and the design acceptance criteria as applicable with IPEC Unit 2 FSAR Section 1.11 for SSE loads. The crane and supporting structure were determined to remain below material yield when subject to the maximum 75 ton load lift combined with the SSE. Standards and guides which have been used for determining allowable stress limits and other acceptance criteria, are consistent with industry practice for similar applications. These include the American Institute of Steel Construction (AISC) Manual, 9th Edition, the American Concrete Institute (ACI) 318-02, Building Code Requirements for Reinforced Concrete and the American Society of Mechanical Engineers (ASME) NOG-1, "Rules for Construction of Overhead Cranes."

The second paragraph of Section 4.5 "Crane Seismic Qualification" of Attachment 1 of the LAR, states that "The crane and supporting structure were determined to remain below yield when subject to the maximum load lift combined with the SSE ...". NRC staff notes that, although this criterion is acceptable for structural steel members when buckling limit states do not govern, the criterion is not appropriate for the structural steel members for which buckling considerations govern the design. Therefore, by letter dated September 7, 2007 (ML072480497), the NRC staff requested additional information regarding the acceptance criteria and maximum force/stress levels in the crane seismic qualification as follows:

- Clarify the acceptance criteria used for structural steel members of the crane, the wire ropes and other important load carrying components of the FHB crane system considering the governing failure limit states.

- List the maximum force/stress levels in the important members/components of the crane and its supporting structure under the critical load combination with seismic SSE loading, and the corresponding acceptance criteria with basis, and the factors of safety.

- Describe the factor of safety provided in the design/selection of lifting devices attached to the load block.

The licensee summarized the stress levels, for the applicable load combinations, in all important crane components: wire rope, trolley, bridge girders, girder bolts, end trucks, rail clamps and bolting, building columns and the column footings. The maximum ratio of actual stress to the allowable for any component was 0.96, which occurred in the crane bridge girders.

In an October 3, 2007, response letter (ML073050247), the licensee stated that the acceptance criteria for the various crane elements and building structure were based on ASME NOG-1, Section NOG-4300 or AISC. The acceptance criteria for the hoisting wire ropes were based on

CMAA-70 and the fabricated lift yoke was designed and tested to meet the requirements of ANSI N14.6-1993. The licensee presented a summary of results of stress evaluation of all the crane components. The ratios of the actual stress/force to the allowable were in all cases less than 1.0 with a reasonable margin. The staff finds that the acceptance criteria and standards used by the licensee for evaluation of the crane are consistent with standard industry practice and meets the intent of NUREG-0612. Therefore, the response is acceptable.

4.2.4 Tornado Wind and Severe Weather Evaluations

In Section 4.6 of the LAR, the licensee addressed tornado wind and severe weather issues during cask loading operations using the FHB crane. The Unit 1 FHB 75-ton crane is totally contained within the IP-1 FHB. The stack up of the transfer cask and MPC on the HI-STORM overpack will be accomplished using the Unit 2 gantry crane inside the IP-2 FHB. The licensee stated that it currently has administrative controls in place that prohibit IP-2 fuel handling and the FHB doors from being opened if severe weather is imminent. If fuel handling is in progress, current procedures require fuel handling and radioactive material transport to cease except as required to move material to a safe location. The licensee stated that IP-1 cask loading procedures will be verified to contain similar severe weather restrictions.

By letter dated September 7, 2007 (ML072480497), the NRC staff requested additional information with regard to actions taken for severe weather conditions during cask handling operations using the IP-1 FHB crane as follows:

- Describe procedures and administrative controls to be followed prior to commencement of each cask loading operations in the IP-1 FHB building to ensure that fuel handling is stopped and the FHB doors are closed in the event of imminent severe weather.

- Describe what actions will be taken if severe weather becomes imminent after a cask loading operation using the crane has commenced.

The licensee's October 3, 2007, response (ML073050247) described the site specific cask loading procedures that will instruct cask loading personnel to contact the Control Room prior to initiating cask movement activities to verify that severe weather is not imminent. Their response also indicated the transfer cask is moved into the FHB on air pads and the exterior door will be closed prior to any crane-related lifting activities, and that if severe weather becomes imminent after a cask lift has commenced, the load will be placed in a safe condition either back on the cask load pool floor slab or on the FHB elevation 70 ft slab.

NRC staff finds that the licensee has site specific procedures that would direct actions to verify severe weather is not imminent prior to commencing cask movement activities and a plan of action to place the load in a safe condition if severe weather becomes imminent after a cask lift has commenced. Therefore, the response is acceptable.

4.2.5 Conclusion on the Structural Adequacy of the Crane

Based on its review of the licensee's submittal and detailed RAI responses, the staff finds that the licensee performed a comprehensive structural design evaluation (including seismic qualification), inspections of crane structural steel, functional and 125% proof-load testing, and severe weather evaluations of the IP-1 FHB 75-ton crane in support of its use for the proposed spent fuel cask loading operations. The licensee performed an analysis that demonstrated that the crane is structurally adequate to handle the rated load under the limiting loading conditions including a seismic SSE event concurrent with a loaded cask suspended from the main hook. These analyses/evaluations, inspections and tests were properly performed using methods that are consistent with industry standards and standard practices and yielded acceptable results that meet the intent of NUREG-0612. The IP-1 FHB 75-ton crane is thus adequately designed and appropriately maintained, inspected, and tested to provide reasonable assurance that the cask operation loads can be safely handled without a load drop. Therefore, the staff finds that the use of the FHB 75-ton crane for the proposed spent fuel dry cask handling operations at IP-1 is acceptable.

4.3 Load Drop Analysis of the Spent Fuel Cask and Components

The staff has reviewed the licensee's technical and regulatory analyses in support of the proposed license amendment, which are described in Sections 4.0 and 5.0, respectively, of the applicants LAR that was submitted as Attachment 1 to its February 22, 2007, letter. The cask loading operation proposed by the licensee involves the considerations regarding NUREG-0612. Specifically, the cask loading operation involves handling and control of heavy loads inside the FHB. As such, considerations are given to the design and operation of the FHB crane, the proposed movement of the transfer cask, the use of procedures for loading and handling, and analyses of potential load drops. A matrix showing the degree of compliance with the guidelines prescribed in NUREG-0612 was provided as Table 4 of the licensee's February 22, 2007, letter. The license amendment proposes to change the plant's FSAR/Decommissioning Plan to reflect the use of the FHB crane for dry spent fuel cask component lifting and handling operations.

4.3.1 Fuel Handling Building Crane/Loading Operations

The HI-STORM 100S Version B, Type 185 system will be used for dry cask storage for nuclear fuel at the IP-1 ISFSI. The FHB crane will be used to lift and handle the HI-TRAC 100D Version IP1 transfer cask and MPC-32, a 32-assembly PWR fuel storage MPC. The combined lift weight, including rigging and lift yoke, to be handled during cask loading operations will not exceed 75-tons.

During the cask loading evolution, individual spent fuel assemblies are moved from the spent fuel racks in the west pool, through a gate into the disassembly pool, through a second gate into the cask load pool and loaded into the MPC, which is inside the HI-TRAC transfer cask in the cask load pool. Once the 32 pre-designated fuel assemblies have been loaded into the MPC, the MPC lid is installed underwater and the transfer cask is lifted by the FHB crane and placed on the truck bay floor in the cask preparation work station area. In this area, the canister is welded, drained, dried, and backfilled with helium. The transfer cask containing the sealed MPC is lifted a few inches by the FHB crane and placed on an air castor to be moved northward out of the FHB. This concludes the lifting operations by the Indian Point Unit 1 FHB crane in support of dry spent fuel storage. Lifting evolutions performed by cranes other than the Indian Point Unit 1 FHB crane have not been reviewed or approved by the NRC in this safety evaluation report.

4.3.2 NUREG-0612 General Guidelines

In Table 4 of its February 22, 2007, letter (ML070740552), the licensee provided a matrix comparing the FHB crane design with the applicable regulatory guidelines in NUREG-0612.

Included in the evaluation column of this matrix is a discussion on how the objectives and general guidelines of Section 5.1.1 of NUREG-0612 will be met with regards to: (1) use of defined safe load paths, (2) specific procedure development, (3) training and qualification of crane operators, (4) use of special lifting devices, (5) installation and use of non-custom lifting devices (6) inspection, testing, and maintenance of cranes, and (7) application of codes and standards to crane design.

With regard to NUREG-0612 Section 5.1.1(1), safe load paths, the licensee states that safe load paths for heavy load movements will be defined within a specific cask lifting procedure for Indian Point Unit 1 and that the transfer cask is prevented from traveling over the reactor vessel and the spent fuel pool due to the pool configuration. The licensee also states that the safe load path does not result in any lifts above or near any safe shutdown equipment. In Attachment 3, the licensee has committed to defining the safe load path in specific cask loading procedures and clearly marking it on the floor prior to the first lift of the loaded transfer cask.

NUREG-0612 Section 5.1.1(2) gives guidance relating to procedural development to cover load-handling operations for heavy loads that are or could be handled over or in proximity to irradiated fuel. The licensee states that it will prepare specific crane operating procedures covering the areas discussed in NUREG-0612 Section 5.1.1(2). These include: identification of required equipment, inspections and acceptance criteria required before load movement, the steps and proper sequence to be followed in handling the load, defining the safe load path, and other special precautions. The licensee also states that additional details for controlling movement during transfer cask handling operations will be provided in a specific cask loading and handling procedure. A commitment regarding these procedural developments has been made in Attachment 3.

To satisfy the guidelines of NUREG-0612 Section 5.1.1(3), the licensee indicated that crane operators are trained in the area of material handling and hoisting and equipment control and that the training includes Chapter 2-3 American National Standards Institute (ANSI) B30.2-1976, “Overhead And Gentry Cranes.” This has been committed to in Attachment 3.

Guidance on the use of special lifting devices is provided in NUREG-0612 Section 5.1.1(4). The licensee states that the only device required to meet the guidelines for use of special lifting devices provided in NUREG-0612 Section 5.1.1(4) is the Holtec HI-TRAC lifting yoke. According to the licensee, the lifting yoke complies with the guidelines of ANSI-N14.6-1993, “Special Lifting Devices For Shipping Containers Weighing 10,000 Pounds Or More For Radioactive Material,” and the additional guidelines of NUREG-0612 Section 5.1.6(1)(a). The licensee also states that the lifting trunions of the HI-TRAC cask have a design safety factor that is greater than 10 times the maximum combined static and dynamic load.

NUREG-0612 Section 5.1.1(5), installation and use of non-custom lifting devices, does not apply since the licensee states that transfer cask lift and transfer to the loading floor does not utilize any lifting devices that are not specifically designed.

To ensure the guidelines of NUREG-0612 Section 5.1.1(6), inspection, testing, and maintenance of cranes, are met, the licensee confirms that the FHB crane is inspected, tested, and maintained in accordance with Chapter 2-2 of ANSI B30.2-1976 and the additional guidance contained in NUREG-0612. The licensee has completed a full-load proof test on the crane

The FHB crane conformed to ANSI B30.2 and Crane Manufacturers Association of America (CMAA) standards in place at the time of fabrication. Testing and inspections, including a full-load proof test at minimum operating temperature and nondestructive examination of critical welds after the proof test will be performed to offset the material design requirements of the current ANSI B30.2 and CMAA-70, “Specifications For Overhead Traveling Cranes,” guidelines. This satisfies the guidance on application of codes and standards to crane design given in NUREG-0612 Section 5.1.1(7).

Based on the staff review of the information provide by the licensee in its submittal and in the NUREG-0612 comparison matrix for the Indian Point Unit 1 FHB crane, the staff finds that the licensee satisfies the general guidelines given in Section 5.1.1 of NUREG-0612.

4.3.3 NUREG-0612 Spent Fuel Pool Area Guidelines

To provide assurance that the evaluation criteria of Section 5.1 of NUREG-0612 are met for load handling operations in the spent fuel pool area, in addition to satisfying the general guidelines of Section 5.1.1 of NUREG-0612, one of four sets of additional guidelines must be met. These guidelines are provided in Section 5.1.2 of NUREG-0612. Table 4 of the February 22, 2007, letter gives the licensee’s evaluation and compliance with these additional guidelines. Specifically, the licensee conforms with items 2 and 4 of Section 5.1.2 of NUREG-0612.

Item 2 of Section 5.1.2 of NUREG-0612 has 5 sub-guidelines to mitigate the potential damage caused by a load drop event by a non-single-failure-proof crane. Item 2 includes limiting movement of the overhead crane through mechanical and electrical interlocks and limits on lifting height. In regards to NUREG-0612 Section 5.1.2(2)(a), the licensee states that mechanical stops or electrical interlocks will be provided that prevent movement of the overhead crane load block over or within 15 feet (horizontal) of the spent fuel pool. These mechanical stops or electrical interlocks will be controlled by administrative controls. A commitment to this effect has been made in the licensee’s submittal. To satisfy NUREG-0612 Section 5.1.2(2)(b), the licensee states that an electrical interlock will be provided to prevent the bridge from traveling toward the fuel pool when it is moving the transfer cask or its components. Analysis has been performed by the licensee and they have concluded that a dropped cask in the cask load pool will not result in leakage from the spent fuel pool that could uncover the fuel. Guidance in NUREG-0612 Section 5.1.2(2)(c) is given to preclude a dropped cask from rolling. The licensee states that the transfer cask will not be carried more than six inches above the FHB operating floor, which satisfies this guideline. NUREG-0612 Section 5.1.2(2)(d) is met because there is no safety-related or safe shutdown equipment in the vicinity, although mechanical stops or electrical interlocks will be provided to prevent transfer cask movement over the fuel pool. To satisfy NUREG-0612 Section 5.1.2(2)(e), the licensee confirms that cask drop consequence analysis have been performed in accordance with the guidelines of Appendix A of NUREG-0612.

Item 4 of NUREG-0612 Section 5.1.2 requires that the effects of drops of heavy loads be analyzed and shown to satisfy the evaluation criteria of Section 5.1 of NUREG-0612 and conform to the guidelines in Appendix A or NUREG-0612. The licensee states that cask drop analyses have been performed that conform to the guidelines of Appendix A of NUREG-0612.

Based on review of the information provide by the licensee in its submittal and in the NUREG-0612 comparison matrix for the IP-1 FHB crane, the staff finds that the licensee satisfies the additional guidelines given in Section 5.1.2 of NUREG-0612.

4.3.4 Load Drop Considerations

Because the FHB crane is not single-failure-proof crane, the licensee has identified a requirement to perform additional load drop analysis due to a difference in the design of the proposed transfer cask from the existing licensing basis. The regulatory difference is that the transfer cask to be used to support the dry fuel storage initiative (HI-TRAC 100D Version IP1) is not certified to the standards of 10 CFR 71 as was the previous shipping cask (GE IF-200). Certain drops of the transfer cask, MPC, and MPC lid were postulated. The locations of where the drops were postulated and evaluated were chosen to comply with applicable parts of Part 50 licensing requirements, NUREG-0612, and NRC Bulletin 96-02.

To mitigate the consequences of two of the postulated drops, the licensee will use an impact limiter in the cask load pool where the transfer cask will be moved in the vertical direction. In attachment three of the February 22, 2007, letter, the licensee has committed to installing the properly designed impact limiter in the cask loading pool prior to the first lift of the transfer cask loaded with fuel. Although NUREG-0612 states that energy absorbing devices must be attached to the cask during load handling operations for credit to be taken, it is generally accepted that the impact limiter can be separate from the cask if the load path prevents lifts over anything except the impact limiter in the spent fuel pool.

The analyses submitted by the licensee confirmed that the postulated load drops of the 75-ton transfer cask in the cask load pool or on the elevation 70 ft 6 in. concrete floor would not result in significant damage to any safety related structures, systems, or components (SSC) and the cask would experience a deceleration less than the allowable limit for the fuel. The postulated drops of the MPC lid resulted in four MPC-32 IP1 basket cells being damaged, although major relocation of fuel is not expected.

Based on the staff's review of the drop analyses provided by licensee in 4.7.2 of its LAR submittal, the staff finds that the licensee has included the considerations and assumptions stated in NUREG-0612, Appendix A, and conforms to the guidelines.

4.3.5 Conclusion on Load Drop Analysis of the Spent Fuel Cask and Components

Based on the review of the licensees submitted information on the handling of heavy loads associated with this amendment request, the staff finds the licensee has provided adequate assurance that their planned actions for the handling of heavy loads associated with dry cask storage loading operations are consistent with the "defense-in-depth" approach to safety described in NUREG-0612. Therefore, the staff finds the amendment request acceptable for the handling of heavy loads.

4.4 Load Drop Analyses of the Spent Fuel

Because the FHB is not a single-failure-proof crane, certain drops of the transfer cask, MPC, and MPC lid were postulated and evaluated. The drop locations were chosen to comply with applicable portions of 10 CFR Part 50 licensing requirements, and guidance of NUREG-0612

Appendix A, and NRC Bulletin 96-02. The licensee has reviewed the required fuel building cask handling operations and has evaluated drops at various locations along the load path in order to be in compliance with the applicable regulatory requirements and additional guidelines in NUREG-0612.

The HI-STORM 100 storage cask system at IP-1 consists of an MPC, which holds up to 32 PWR fuel assemblies, a HI-TRAC 100D-Version IP1 transfer cask, which contains the MPC during loading, unloading, and transfer operations, and a HI-STAR storage overpack, which provides shielding, heat removal, and structural protection for the MPC deployed at the ISFSI. Attachment 1 to the licensee's February 22, 2007, letter (ML070740552) presents the cask handling operational sequence, summarizes cask loading operations and corresponding load drop scenarios. Additional information is presented in the licensee's October 3, 2007 (ML073050247), and February 27, 2008 (ML080630507), responses. The following subsections provide a discussion of the structural performance of the cask system for withstanding the six most bounding load drops to ensure that the cask system is capable of maintaining its shielding, confinement, and criticality safety functions after the postulated cask drop events.

4.4.1 Loaded Vertical Transfer Cask Drop into the Cask Load Pool

This event is postulated to occur by crane failure at upper limit of lift resulting in drop through air and a continued drop in water onto the impact limiter.

A HI-TRAC transfer cask containing a MPC loaded with 32 fuel assemblies is lifted from the floor of the cask loading pool (Elevation 31 ft 4 in.) to above the cask handling elevation (Elevation 70 ft 6 in.). The cask is dropped vertically and lands upright on the impact limiter installed on the floor of the cask load pool.

The February 22, 2007 (ML070740552), LAR indicates that the simulations were carried out for different impact limiter resistances. The 80% and 120% of nominal strength simulations provide bounding results for standard impact limiter material tolerances. The analysis results demonstrate that the proposed impact limiter configuration successfully maintains an acceptable deceleration level in the transfer cask, MPC, and the contained fuel. The maximum calculated cask deceleration was 35.9 g which is below the 64.8 g limit for the fuel. The maximum impact limiter "crush" was determined to be 13.8 in. The maximum impact force was used to check the bearing capacities of the cask load pool slab and its underlying rock. The minimum safety factor is calculated as 1.78 for the concrete slab. No significant increase in pool leakage is anticipated to result from this drop event since the concrete base slab of the cask load pool remains well within code allowable stress levels. In addition, a permanent weir in each of the connecting gate slots extends to an elevation of 41 ft 3-11/16 in. This weir retains sufficient water shielding in the spent fuel pool to cover the active portion of the fuel while stored in the racks in the event make-up water needed to be added to restore level.

In a September 7, 2007 letter (ML072480497), NRC staff questioned the 64.8 g deceleration limit for fuel based on recent NRC staff evaluations. NRC staff requested the licensee provide additional analysis to demonstrate the IP-1 stainless steel clad fuel will not be damaged in the postulated loaded vertical transfer cask load drop scenario.

The licensee's October 3, 2007 response (ML073050247), provided an analysis to demonstrate that IP-1 fuel is bounded by the spent fuel already considered by NRC staff in NUREG-1864.

The metrics provided in the licensee's submittal indicate that results for the NUREG-1864 Reference fuel will bound results for the IP-1 Analysis Basis fuel (i.e., the IP-1 fuel has lower burnup, has fuel rods of lower total weight, has a larger critical buckling load, and requires a smaller lateral movement before contact with the fixed wall of the storage cell). Therefore, structural integrity of the Analysis Basis IP-1 fuel rods can be directly asserted by comparison with the results from analyses performed in NUREG-1864 using reference fuel it is concluded that:

1. The 4-in. drop on the concrete slab condition is enveloped by the 20-ft drop condition by a large margin.

2. The 40-ft drop on the impact limiter in water will certainly result in more than 10% reduction in strain from the 40-ft drop onto concrete, considering that: (i) the IP-1 fuel will have less than half the kinetic energy, and has less lateral movement space because of the fuel channel around it; and (ii) the impact limiter is sized to absorb virtually all of the kinetic energy.

Therefore, drop scenarios for IP-1 fuel are considered enveloped by those mentioned above in NUREG-1864.

By letter of January 31, 2008 (ML080290311), NRC staff requested the licensee to provide the basis for the 1% failure strain limit for stainless steel cladding for IP-1 spent fuel. In its February 27, 2008 response (ML080630507), the licensee provided a comparison of test results for stainless steel clad fuel which indicates a minimum total elongation of about 5% for fuel with similar operating history to IP-1 spent fuel. Based on this information, NRC staff agrees that the 1% assumption is conservative.

On the basis of the bounded condition of IP-1 fuel by NUREG-1864 reference fuel, the NRC staff finds that: (1) the drop event would not cause fuel damage or fuel relocation that would result in an unanalyzed criticality configuration; and (2) the MPC and transfer cask would retain their structural configurations to permit retrieval of the MPC after the drop.

4.4.2 Inclined Loaded Vertical Transfer Cask Drop into the Cask Load Pool

A HI-TRAC transfer cask containing an MPC loaded with 32 fuel assemblies is lifted from the floor of the cask load pool (Elevation 31 ft 4 in.) to above the cask handling elevation (Elevation 70 ft 6 in.). The transfer cask is dropped and the edge of the transfer cask strikes the impact limiter installed on the floor of the cask load pool.

The simulations were once again carried out for different impact limiter resistances. The 80% and 120% of nominal strength simulations provide bounding results for standard impact limiter material tolerances. The analysis results demonstrate that the proposed impact limiter configuration successfully maintains an acceptable deceleration level in the transfer cask, MPC, and the contained fuel. The maximum calculated cask deceleration was 41.3 g which is below the 64.8 g allowable limit for the fuel. The maximum impact limiter “crush” was determined to be 14.1 in. The maximum impact force was once again used to check the bearing capacities of the cask load pool slab and its underlying rock. The minimum safety factor is calculated as 1.69 for the concrete slab.

In a September 7, 2007 letter (ML072480497), NRC staff questioned the 64.8 g deceleration limit for fuel based on recent NRC staff evaluations. NRC staff requested the licensee provide additional analysis to demonstrate the IP-1 stainless steel clad fuel will not be damaged in the postulated inclined loaded vertical transfer cask load drop scenario.

The licensee's October 3, 2007 response (ML073050247) provided an analysis to demonstrate that IP-1 fuel is bounded by the spent fuel already considered by NRC staff in NUREG-1864. The metrics provided in the licensee's submittal indicate that results for the NUREG-1864 Reference fuel will bound results for the IP-1 Analysis Basis fuel (i.e., the IP-1 fuel has lower burnup, has fuel rods of lower total weight, has a larger critical buckling load, and requires a smaller lateral movement before contact with the fixed wall of the storage cell.) As demonstrated above, the structural integrity of the Analysis Basis IP-1 fuel rods can be assessed by comparison with the results from analyses performed in NUREG-1864 using reference fuel. As in the vertical drop discussed above, it is concluded that drop scenarios for IP-1 fuel are enveloped by those in NUREG-1864.

This finding is further reinforced by the fact that the austenitic stainless steel irradiated to a low fluence in the low flux IP-1 reactor is considerably more ductile than higher burnup Zircaloy fuel.

On the basis of the bounded condition of IP-1 fuel by NUREG-1864 reference fuel, the NRC staff finds that: (1) the drop event would not cause fuel damage or fuel relocation that would result in an unanalyzed criticality configuration; and (2) the MPC and transfer cask would retain their structural configurations to permit retrieval of the MPC after the drop.

4.4.3 Loaded Transfer Cask Tips into the Cask Load Pool.

A HI-TRAC transfer cask containing an MPC loaded with 32 fuel assemblies is lifted out of the cask load pool and begins to traverse horizontally to the east with the cask bottom slightly above elevation 70 ft 6 in. The cask drops just as it begins to travel over the cask load pool east wall and the bottom of the cask strikes the edge of the pool. The cask tilts to the west and rotates with its side contacting the west wall of the cask load pool. The LS-DYNA model consists of the cask initially positioned 18 in. above the east wall curb. Two cases (denoted “Center” and “Edge”) are considered with the only difference being the initial position of the cask relative to the edge of the east wall when the cask impacts. The “Center” case positions the cask pool lid center right over the curb edge to the load pool; while in the “Edge” case the cask has only 1 in. overlap with the east wall edge.

The results of drop simulations of the cask onto the cask load pool east wall top edge with a subsequent rotation of the cask and impact with the cask load pool west wall top edge produces acceptable cask decelerations and only local damage to the wall. Table 1 summarizes the results from simulations of impact with the cask load pool walls. The calculated decelerations are less than the 64.8 g limit for the fuel and are, therefore, acceptable. The 4-ft thick concrete wall only suffers local damage at the impacted region. No significant increase in pool leakage is anticipated as a result of the localized structural damage which could occur as a result of this drop event. Normal pool make-up capability, which can be backed up by emergency make-up from the fire water and city water systems, ensure that no significant loss of shielding will occur for any spent fuel stored in the west spent fuel pool. The design of the interconnecting gates are such that the spent fuel being stored in the west storage pool cannot become uncovered due to a cask drop in the cask load pool. Due to the age of the spent fuel, water inventory is only required for shielding in the vicinity of the fuel pools.

In a September 7, 2007 letter (ML072480497), NRC staff questioned the 64.8 g deceleration limit for fuel based on recent NRC staff evaluations. NRC staff requested the licensee provide additional analysis to demonstrate the IP-1 stainless steel clad fuel will not be damaged in the postulated load drop scenarios.

The licensee's October 3, 2007 response (ML073050247) provided an analysis to demonstrate that IP-1 fuel is bounded by the spent fuel already considered by NRC staff in NUREG-1864. The metrics provided in the licensee's submittal indicate that results for the NUREG-1864 Reference fuel will bound results for the IP-1 Analysis Basis fuel (i.e., the IP-1 fuel has lower burnup, has fuel rods of lower total weight, has a larger critical buckling load, and requires a smaller lateral movement before contact with the fixed wall of the storage cell). Therefore, structural integrity of the Analysis Basis IP-1 fuel rods can be assessed by comparison with the results from analyses performed in NUREG-1864 using the Reference fuel. The staff finds that:

1. The 4-in. drop on the concrete slab condition is enveloped by the 20-ft drop condition by a large margin.

2. The 40-ft drop on the impact limiter in water would certainly result in more than 10% reduction in strain from the 40-ft drop onto concrete, considering that: (i) the IP-1 fuel will have less than half the kinetic energy, and has less lateral movement space because of the fuel channel around it; and (ii) the impact limiter is sized to absorb virtually all of the kinetic energy.

Therefore, drop scenarios for IP-1 fuel are considered enveloped by those mentioned above in NUREG-1864.

On the basis of the bounded condition of IP-1 fuel by NUREG-1864 reference fuel, the NRC staff finds that: (1) the drop event would not cause fuel damage or fuel relocation that would result in an unanalyzed criticality configuration; and (2) the MPC and transfer cask would retain their structural configurations to permit retrieval of the MPC after the drop.

4.4.4 MPC Lid Drop onto the MPC

After the spent fuel is loaded into the MPC, the MPC lid is installed using the FHB 75-ton crane. The rigging, attached to four symmetrically located lift points in the lid, ensures that the lid is held in the horizontal position during lowering to assure fit up onto the MPC. If the lid is dropped from a significant height, the column of water below the falling lid will eventually cause the lid to drift laterally and not physically be able to enter the open MPC. This analysis assumes the lid is 3 ft above the MPC and perfectly positioned for insertion when the postulated failure occurs, allowing the lid to drop straight down into the MPC fuel cavity in the horizontal orientation. The lid will accelerate as it falls and impart an impact load on the four lift lugs welded to the inside of the MPC shell. The lift lugs are designed to support the dead weight of the lid until the lid is welded to the canister shell. The analysis evaluates the ability of the lift lugs to withstand the impact load of the lid drop using manual structural mechanics computational techniques. The acceptance criterion for this analysis is no damage to the spent fuel assemblies in the MPC.

The results of the analysis show that the lift lugs would withstand the impact force and prevent the lid from coming into contact with the fuel basket or the fuel.

In a September 7, 2007 letter (ML072480497), NRC staff requested the licensee to provide the basis for the simplifying assumption that the water's change in density is proportional to the lid velocity, and to provide an analysis of how this assumption affects the analysis. The licensee's October 3, 2007 response (ML073050247) provided an analysis demonstrating how water compressibility was conservatively modeled in this postulated event. Additionally, language of the key assumption under question was reworded to clarify the relationship between water density, fluid velocity and lid velocity.

In consideration of the additional information provided by the licensee in the October 3, 2007 response, and the February 27, 2008 response discussed below, the NRC staff finds that: (1) the drop event would not cause fuel damage or fuel relocation that would result in an unanalyzed criticality configuration; and (2) the MPC and transfer cask would retain their structural configurations to permit retrieval of the MPC after the drop.

4.4.5 MPC Lid Drop onto the Transfer Cask Flange

A second lid drop event was analyzed where the MPC lid is assumed to drop vertically from the elevation of 90 in. above the cask loading pool water surface. The lid is slightly offset from the center of the MPC cavity so that the first impact will occur on the top flange of the HI-TRAC transfer cask. Subsequent to the initial impact, the lid may hit the MPC shell, the fuel basket and the stored fuel assemblies before it eventually rests in the MPC cavity. The objective of the analysis performed, using the LS-DYNA finite element code, is to determine whether the postulated offset drop scenario would significantly damage the fuel basket in the MPC resulting in an unacceptable criticality configuration for the stored fuel assemblies.

The results of the analysis conclude that the MPC lid hits the transfer cask top flange at a velocity of 220.0 in/sec. and that a dropped MPC 32 IP-1 lid could damage up to 4 MPC-32 IP1 basket cells. The total of 4 basket cells suffer plastic deformation extending down as much as 6.3 in. below the basket top, which is inside the region covered by the neutron absorber (3.75 in. below the basket top). The cell deformation is limited to an area more than 12 in. above the active fuel region of the stored fuel. Major relocation of fuel material is not expected. The analysis did not credit the IP-1 fuel loading arrangement which places all fuel assemblies in damaged fuel containers. These containers provide additional assurance that relocation of fuel material leading to criticality is precluded.

By letter of January 31, 2008 (ML080290311), NRC staff requested the licensee to provide an analysis and evaluation of consequences for this postulated load drop that conforms to the guidance of NUREG-0612, Appendix A. NUREG-0612 stipulates that the load drop analyses must consider a load drop in the orientation causing the most severe consequences, and must also consider the maximum damage that could result. The submittal considers a lid drop in a near horizontal orientation, versus a vertical orientation. In its February 27, 2008, response (ML080630507), the licensee provided a rationale supporting the near horizontal orientation, based on the relative size of the lid to the drop distance, the stability of the orientation due to large moment of inertia of the axis perpendicular to the lid, and the lid lift yoke and rigging design. The response also indicated that their analysis considered damage to all 32 loaded spent fuel assemblies. A vertically oriented cask lid drop would only be expected to impact 2 rows of spent fuel, or about 12 to 14 assemblies.

Based on the licensee's analyses, the NRC finds that, while the cask lid drop in a nearly horizontal orientation is not the most damaging, the licensee's damage analysis did evaluate the most severe consequences, and therefore the staff agrees that: (1) the drop event would not cause fuel damage or fuel relocation that would result in an unanalyzed criticality configuration; and (2) the MPC and transfer cask would retain their structural configurations to permit retrieval of the MPC after the drop.

4.4.6 Loaded Transfer Cask and Vertical Drop onto the Elevation 70 ft 6 in. Concrete Floor**.**

This event would occur after the loaded transfer cask is lifted out of the cask load pool and is traversing to the east to the cask preparation area for canister welding and other activities. This analysis fixes the allowable height above the floor to which the loaded transfer cask can be lifted while being moved. This also bounds placement of the loaded transfer cask on the air transporter pads.

By administratively limiting the carrying height, when the cask is above the spent fuel building floor level, the cask would experience a maximum deceleration below the 64.8 g allowable limit for the fuel.

In a September 7, 2007 letter (ML072480497), NRC staff questioned the 64.8 g deceleration limit for fuel based on recent NRC staff evaluations. NRC staff requested the licensee provide additional analysis to demonstrate the IP-1 stainless steel clad fuel will not be damaged in the postulated load drop scenarios.

The licensee's October 3, 2007 response (ML073050247) provided an analysis to demonstrate that IP-1 fuel is bounded by the spent fuel already considered by NRC staff in NUREG-1864. The metrics provided in the licensee's submittal indicate that results for the NUREG-1864 Reference fuel will bound results for the IP-1 Analysis Basis fuel (i.e., the IP-1 fuel has lower burnup, has fuel rods of lower total weight, has a larger critical buckling load, and requires a smaller lateral movement before contact with the fixed wall of the storage cell). Therefore, structural integrity of the Analysis Basis IP-1 fuel rods can be assessed by comparison with the results from analyses performed in NUREG-1864 using the Reference fuel. The staff finds that:

1. The 4-in. drop on the concrete slab condition is enveloped by the 20-ft drop condition by a large margin.

2. The 40-ft drop on the impact limiter in water will certainly result in more than 10% reduction in strain from the 40-ft drop onto concrete, considering that: (i) the IP-1 fuel will have less than half the kinetic energy, and has less lateral movement space because of the fuel channel around it; and (ii) the impact limiter is sized to absorb virtually all of the kinetic energy.

Therefore, drop scenarios for IP-1 fuel are considered enveloped by those mentioned above in NUREG-1864.

On the basis of the bounded condition of IP-1 fuel by NUREG-1864 reference fuel, the NRC staff finds that: (1) the drop event would not cause fuel damage or fuel relocation that would result in an unanalyzed criticality configuration; and (2) the MPC and transfer cask would retain their structural configurations to permit retrieval of the MPC after the drop.

4.4.7 Conclusion of Load Drop Analyses of the Spent Fuel

Based on its review of the licensee's submittal and detailed RAI responses, the staff finds that the licensee performed comprehensive load evaluations for a postulated loaded vertical transfer cask drop into the cask load pool, an inclined loaded vertical transfer cask drop into the cask load pool, a loaded transfer cask tipping into the cask load pool, an MPC lid drop onto the MPC, an MPC lid drop onto the transfer cask flange, and a loaded transfer cask and vertical drop onto the concrete floor. The licensee performed analyses that demonstrated for these scenarios that: (1) the drop event would not cause fuel damage or fuel relocation that would result in an unanalyzed criticality configuration; and (2) the MPC and transfer cask would retain their structural configurations to permit retrieval of the MPC after the drop. These analyses/evaluations, inspections and tests were properly performed using methods that are consistent with industry standards and standard practices and yielded acceptable results that meet the intent of NUREG-0612. Therefore, the staff finds that the use of the FHB 75-ton crane for the proposed spent fuel dry cask handling operations at IP-1 is acceptable.

5.0 SUMMARY

The changes proposed by this LAR will authorize the licensee to allow use of the FHB crane in support of dry storage of spent nuclear fuel at IP-1.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the appropriate New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendment allows spent fuel stored in the Indian Point Unit 1 FHB to be moved into dry storage utilizing the 75-ton FHB crane.

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding published on June 19, 2007, (72 FR 33779). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

8.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: 1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; and 2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security nor to the health and safety of the public.

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