U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-286/88-16

Docket No. 50-286

License No. DPR-64 Category C

Licensee: New York Power Authority
P. O. Box 215
Buchanan, New York 10511

Facility Name: Indian Point 3

Inspection At: Buchanan, NY

Inspection Conducted: August 16-19, 1988

Inspectors: P. O'Connell — 9/15/88
P. O'Connell, Radiation Specialist, FRPS — date

S. Sherbini — 9/2/88
S. Sherbini, Senior Radiation Specialist, FRPS — date

Approved by: M. Shanbaky — 9/20/88
M. Shanbaky, Chief, Facilities Radiation Protection Section, FRSSB, DRSS — date

Inspection Summary: Inspection conducted on August 16-19, 1988 (Inspection Report No. 50-286/88-16)

Areas Inspected: Routine, announced, inspection to review the status of outstanding items identified during previous inspections, and to review the status of the steam generator replacement project.

Results: No violations were identified.

DETAILS

1.0 Personnel Contacted

1.1 Licensee Personnel

* R. Deschamps, Health Physics General Supervisor
* J. Gillen, Chemistry General Supervisor
M. Lewis, Consultant for Dosimetry Services
D. Mayer, ALARA Radiological Engineer
* J. Perrotta, Rad. & Env. Services Superintendent
* D. Quinn, Supervisory Radiological Engineer
* J. Russell, Superintendent of Power
* J. Schivera, Licensing Coordinator

1.2 NRC Personnel

* G. Hunges, Resident Inspector

2.0 Status of Previously Identified Items

2.1 Concern in Inspection Report 87-02: The concern was that the technical personnel associated with the dosimetry program were involved in the daily routine of the dosimetry operation and had no time to devote to oversight of the program, which included review and improvement of procedures, upgrade of equipment, and oversight of the quality control program.

The licensee stated that a radiological engineer now provides the necessary technical oversight for the dosimetry program. In addition, a dosimetry technical consultant has been hired to assist in re-writing the dosimetry procedures and writing additional procedures for newly aquired systems. The staffing level currently includes three dosimetry clerks and a dosimetry supervisor. The clerks operate the dosimetry equipment, enter the data into the dosimetry computer data system, and issue the dosimeters. The supervisor reports to the Supervisory Radiological Engineer. She supervises the dosimetry operation and administers the quality control program, and prepares a quarterly quality control report. The licensee stated that a monthly report is to be instituted soon that will include the QC data as well as a review of any dosimetry problems encountered during the period. Technical oversight is normally provided by the Dosimetry Radiological Engineer. Currently, this engineer is involved in a special project, and the technical oversight is being provided by the ALARA radiological engineer. The radiological engineer reviews the QC data and signs the reports. The Dosimetry Supervisor is being provided with training in dosimetry to upgrade her technical understanding of dosimetry principles and of the system hardware. The licensee stated that the dosimetry clerk staff is augmented during outages. Operational health physics maintains and

issues the self reading dosimeters, and also irradiates the TLD badges when required. The Dosimetry department handles the self-reading dosimeter (SRD) data and also investigates significant SRD/TLD discrepancies.
Based on the above data, the staffing arrangements in the dosimetry section appear to be satisfactory.

2.2 Concern in Inspection Report 87-02: This item concerned the quality of the dosimetry procedures. The procedures were found to be incomplete, not well written, and did not address many aspects of the dosimetry operation. The licensee has since hired a consultant for the dosimetry services section. The consultant has reviewed the existing procedures and the dosimetry operation and has re-written all the procedures and also added procedures for the areas that were not previously addressed. The new procedures have corrected the deficiencies noted in this item of concern, and they represent a marked improvement in quality. The inspector noted, however, that there were still some parts of the procedures that were not clearly written and some quantities that were not clearly defined. The licensee stated that the procedures are used in conjunction with a training program that serves to clarify the items. This concern is therefore considered resolved.

2.3 (Closed) Unresolved Item (87-02-01): This unresolved item concerned the ability of the licensee to measure beta dose using their TLD dosimeter, which is their dosimeter of record. The dosimeter element that is used to measure the skin dose is covered with a window about 0.018 cm thick. The licensee used the reading of this element to quantify the skin dose without making appropriate corrections to allow for the fact that the skin dose is to be measured at a depth of 0.007 cm. This practice was shown to result in underestimation of the skin dose by a factor of two or more. The licensee has modified their beta protection and skin dose measurement methods to correct this situation. A beta protection policy has been developed and criteria have been established to determine when beta protection is required. If the beta protection criteria are met, the workers are provided with protective clothing and face shields of sufficient thickness to protect all exposed areas of the skin from the beta radiation. The interpretation of the TLD data remains unchanged; that is, the licensee still does not use correction factors to allow for the thickness of the filter over the beta element of the dosimeter. However, based on experimental data, the licensee has modified the method of placement of the TLD on the worker to compensate for this effect. The new policy requires that the dosimeter be worn outside the coveralls if one layer of protective clothing is worn. If two layers are worn the TLD is placed outside the inner layer and inside the outer layer. The data presented by the licensee shows that the dosimeter used in this manner provides a good measure of the skin dose for energies down to a beta endpoint of 0.7 MeV or lower. This method of dosimeter placement, coupled with the beta protection policy, appears to be an acceptable beta protection and measurement policy. The inspector stated that it is necessary to incorporate the details of this policy, as well as its limitations, into the site procedures as well as the radiation protection plan. The licensee stated that this will be completed soon. Inclusion of the policy into the procedures and the plan will be reviewed during a future inspection.

2.4 (Closed) Unresolved Item (87-18-01): This item addressed two issues. One of the issues was the method used to calculate the skin dose resulting from a hot particle contamination on March 31, 1988. At the time of the inspection, there was uncertainty regarding the method used to calculate the skin dose, and the licensee was in the process of making additional measurements and calculations. The second issue in the unresolved item addressed weaknesses in the licensee's program for control and detection of hot particles.

A review of the licensee's method for assessing the skin dose resulting from the incident on March 31 showed that the method is acceptable. The dose was assessed by measuring the beta energy spectrum emitted from the hot particle that was retrieved from the contamination incident. The measurement was made at the Idaho National Engineering Laboratory using a plastic scintillation spectrometer. The beta spectrum is multiplied by a dose rate conversion function that converts the number of beta particles of specified energy incident on the surface of the skin to a dose rate to the skin at a depth of 0.007 cm. The dose rate conversion function was derived by comparing the spectrometer results with the dose rates measured for several beta sources using an extrapolation chamber.

The licensee stated that they have instituted several changes in their hot particle program to address the concerns identified in this unresolved item. A review of the changes shows that these concerns have been addressed. The changes included the following:

- Changes in the appropriate procedures to incorporate specific hot particle items.
- Incorporation of hot particle issues in radiation worker training.
- Hot particle training for the radiological controls technicians.
- Special surveys for hot particles in areas of possible hot particle contamination.
- Audits of the laundry vendor and random frisking of returned laundry. Also microdosimeter surveys of clean protective clothing storage bins.
- Source term reduction, including checks for defective fuel elements, use of low cobalt components, and improved valve packings.
- Addition of a flow chart in the contamination procedures to assist the technician in deciding whether a hot particle is present and to make quick field dose calculations.
- Contact with other utilities for their experience, and attend industry seminars and courses on the subject.

2.5 Concern Identified in Inspection Report 88-02: The concern identified in the inspection report was that, although audits of program activities were being conducted at the required frequency by the QA department, they were being conducted by personnel with no expertise in the area of health physics or radiation protection. The audits were thus not capable of identifying program weaknesses. Furthermore, although the Radiation Protection group conducts a program of self assessment to periodically review certain activities, the results of these reviews were not officially documented. This lack of official documentation weakens the program because it does not provide for follow-up of corrective actions on a systematic basis.

Discussions with the licensee indicated that corrective actions to address these concerns are in progress but have not yet been completed. This item will therefore be reviewed during a future inspection.

2.6 Concern Identified in Inspection Report 88-02: The licensee had committed to complete revision of all the radiological protection procedures by 31 August 1988. The procedures were still in revision at the time this concern was raised. A review of the status of the procedure revision project showed that the procedures have been renumbered to correspond to the organization of the newly issued radiation protection plan. A majority of the procedures have been rewritten, consolidated, and generally improved. The procedures are expected to be issued by the end of August, pending final review and approval. This item will be reviewed during a future inspection.

2.7 Concern Identified in Inspection Report 88-02: During the inspection for the 88-02 inspection report, the inspector noted that there were no post-job reviews for jobs done during the previous outage. There were also no job packages for completed jobs. Discussions with the licensee revealed that the post-job reviews had been done and were available for inspection. The reviews had not been compiled and ready during the previous inspection. The licensee stated that they were still working on assembling job packages for completed jobs. The licensee stated that all the paperwork for completed jobs was available and was readily accessible, but they were not assembled in a complete package for each job. This item will therefore be reviewed during a future inspection.

3.0 Status of the Steam generator Replacement Project: The steam generator (SG) replacement project involves replacing all four SGs by improved but functionally identical Westinghouse SGs. The replacement was prompted by the current state of plugging in the existing generators (average of 24% of the tubes plugged) and the expense, both financial and radiation exposure, of anticipated tube sleeving and repair work on cracks that developed in the girth welds in the existing generators. The new generators are expected to improve ALARA performance on site because of the use of low cobalt components and other improved design features. The replacement is expected to take place as part of the outage scheduled to start in February 1989. The new steam generators are to arrive on barges up the Hudson River, and are due on site in November 1988. The old steam generators are to be stored on site for an indefinite period in a specially constructed building. This building was near completion at the time of this inspection. The walls are 3.5 foot concrete with a concrete ceiling, and the building is to be sealed except for a single access doorway provided to allow for periodic (quarterly) surveys of the stored generators.

Before starting the project, there will be a general containment decontamination. Most exposed surfaces around work areas will be cleaned. An ongoing decontamination program will also be maintained during the project. The reactor will be completely defueled, the cavity will be drained and decontaminated, and the reactor internals will be replaced in the reactor. A 15" concrete shield will then be placed in the cavity, over the reactor vessel, to reduce dose rates during work on the refueling floor. The reactor head will remain in its shielded laydown area on the refueling floor. Shielding in the form of lead blankets will also be used to shield piping and equipment such as the reactor coolant drain tank and the regenerative heat exchanger. Temporary shielding will also be installed around the reactor coolant pipes close to the areas where the cuts will be made. A platform will be constructed to cover the reactor cavity, with the top of the platform level with the refueling floor. The platform will support the steam generators on their way to and from the equipment hatch.

The old SGs will not be decontaminated before removal. They will be disconnected from the system by cutting the coolant pipes at the nozzles located at the channel head. A sheet metal enclosure will be erected around the cutting area and will be provided with a ventilation system with roughing and HEPA filters. This system will discharge into the containment exhaust system. A similar ventilation system will be installed on the open SG manway. The containment will be maintained at a slight negative pressure by the containment purge exhaust system, which is normal practice during outages. Heaters will be provided to warm incoming air during the winter months. Cuts other than those at the channel head nozzles will not require any special radiological precautions. The secondary side of the steam generator will be filled with water to provide shielding during the cutting operations. After the cuts are made, the SG will be lifted to the refueling floor level and the nozzles will be shut by welding steel plates over the openings. Steel plates will also be welded over the main steam and feed nozzles. Other smaller piping connections, such as instrument taps, drain piping, etc, will also be welded shut or plugged. The exposure rate from the SG at the time it is ready for moving out of the equipment hatch is expected to be less than 200 mR/hr on contact. The SG will be removed through the equipment hatch in the horizontal position. It will first be lifted from its location to the refueling floor (95′ elevation) and then moved over the platform covering the reactor cavity. In order to make this move, the SG must clear the bioshield at that elevation. This is not possible, and therefore part of the bioshield will be cut off to allow this maneuver. The shield to be cut will be decontaminated first, and the cutting process will be a wire cutting method with a water spray used to reduce dust. No other precautions are expected to be needed. Items to be temporarily stored outside containment will either be decontaminated or bagged and then stored in the interim radwaste storage facility on site.

A new containment access facility is being constructed to allow for the expected large numbers of people entering containment during this project. This will allow people to be processed directly into containment. A new outage support building is also being constructed. The licensee stated that a new computerized access system will be in use at the time the outage starts. This is expected to make access control more efficient and more reliable.

The licensee stated that radiological control during the replacement project will be handled by the site Radiological Control department. This group is also involved in planning for the project and in review of all relevant material provided by the contractor in charge of designing the project, including sign-off on drawings, procedures, etc. The contractor is Bechtel Corporation. Mockup training for special jobs is to be provided by the vendors for these jobs, under the supervision of the contractor and the licensee. A mockup facility will also be erected on site for additional mockup training. The total exposure for the replacement project has been estimated at about 1150 man-rem. Of these, preparations are expected to consume 250 man-rems, removal of the SGs 457 man-rem, installation of the new SGs about 280 man-rem, and post-installation work about 160 man-rem. The licensee stated that this estimate may go up to about 1700 man-rem if they encounter problems in matching the nozzles in the new SGs with the existing coolant piping. In this case additional pipe cutting will have to be done to effect a match. The estimates were based on radiation field data provided by the licensee's site Radiological Controls group. The licensee stated that these estimates do not include any work that is normally done during routine outages.

4.0 Exit Meeting

The inspector met with licensee representatives at the end of the inspection on August 19, 1988. The inspector summarized the purpose, scope, and findings of the inspection.