U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-354/94-06

Docket No. 50-354

License No. NPF-57

Licensee: Public Service Electric and Gas Company
P. O. Box 236
Hancocks Bridge, New Jersey

Facility Name: Hope Creek Nuclear Generating Station

Inspection At: Hancocks Bridge, New Jersey

Inspection Conducted: March 14 - 18, 1994

Inspector: J. Noggle 4/5/94
J. Noggle, Senior Radiation Specialist date

Approved by: Joseph D... 4/6/94
for R. Bores, Chief, Facilities Radiation Protection Section date

Areas Inspected: Areas covered in this inspection included personnel exposure control and exposure reduction techniques employed during the fifth refueling outage at the Hope Creek Station.

Results: The licensee's radiation protection organization performed effectively in providing the necessary radiation safety oversight of outage activities. However, opportunities for in the ALARA program were noted by the inspector. No safety concerns or violations of NRC regulatory requirements were identified during the inspection.

DETAILS

1.0 Individuals Contacted

1.1 Public Service Electric and Gas Company

J. Barber, Hope Creek Radiation Protection Technician, RHR Crosstie Modification
R. Bommer, Hope Creek ALARA Supervisor
* V. Cialante, Hope Creek Senior Radiation Protection Supervisor, ALARA
G. Dodge, Hope Creek Lead Radiation Protection Technician, Torus
* R. Gary, Hope Creek Senior Radiation Protection Supervisor, Operations
* M. Gray, Hope Creek Licensing Engineer
A. Hoornik, Hope Creek Chemist
* J. Molner, Hope Creek Senior Radiation Protection Supervisor
* M. Prystupa, Hope Creek Radiation Protection Manager
J. Seymore, Hope Creek Lead Radiation Protection Technician, Drywell
S. Szymanski, Hope Creek Radiation Protection Supervisor, Balance of Plant
T. Wallender, Hope Creek Radiation Protection Supervisor, Drywell
* J. Wray, Salem Radiation Protection Principal Engineer

1.2 USNRC Personnel

* C. Marschall, Senior Resident Inspector

* Denotes attendance at the exit meeting on March 18, 1994.

2.0 Purpose

This inspection was an announced safety inspection of the Hope Creek Nuclear Generating Station radiation control program during the fifth refueling and maintenance outage.

3.0 Organization

The licensee's radiation protection organization normally consists of approximately 50 personnel. During this outage an additional 73 contract HP personnel were hired. The HP staff was supervised by 11 PSE&G Hope Creek HP supervisors. In addition, three PSE&G Salem HP supervisors provided 24-hours/day leadership over the contract HP technician workforce. Adequate manpower resources were available to accomplish the outage activities in a safe manner.

4.0 Training and Qualifications

The inspector reviewed selected contractor resumes and training qualifications to ensure the minimum requirements of ANSI 3.1-1981 were met for HP technicians responsible for job coverage and the radiological protection of the workers. In all cases, the criteria were met. The experience levels ranged between 3 and 10 years with a median experience level of 7 years. This compares favorably with the requirement for 3 years experience.

The contractor HP training program was accomplished in one week. The HP technicians were given an initial HP screening examination followed by three days of general employee processing. A reading list of required procedures was provided followed by a seven-hour site specific briefing class and a 50-question multiple-choice examination. The inspector reviewed the examination and noted that the test evaluated only a sampling of the procedural areas. The licensee also provided a two-hour practical factors evaluation of each HP technician. The practical factors evaluation was a non-specific oral board evaluation given several weeks earlier that could not be reviewed by the inspector. In general, the contractor HP technician training appeared to be adequate.

5.0 Radiological Occurrence Reports

The inspector reviewed the licensee's radiological occurrence reports (RORs) to evaluate management oversight of radiological events at the station. The licensee classified RORs into three categories, with a level-one ROR indicating least severe and a level-three ROR as the most severe (e.g., contamination found outside the RCA or internal exposure >50 mrem). During 1993, the licensee recorded 37 RORs with five classified as level-3 events. Approximately half of the RORs were personnel contamination events. "Work practices less than adequate" was listed as the most common cause of all RORs. During January and February of 1994, a total of 14 RORs were documented with three rated as level-3 events. During the first 11 days of the fifth refueling outage, 16 RORs were documented. Three RORs involved alarming electronic dosimeters associated with workers signed in under incorrect radiation work permits (RWPs). No job conflict or exposure risk was incurred in any of these cases, and the licensee handled the problems expeditiously with appropriate corrective actions. All other outage RORs were of lower safety significance.

In general, the licensee has a good low threshold for writing RORs and the frequency of safety significant radiological events has been low. Management oversight of radiological problems was effective.

6.0 Exposure Reduction

The inspector reviewed with the licensee the efforts made to reduce the station radioactive source term through injection of zinc into the feedwater system. Zinc injection is used based on the theory that it would become incorporated into the corrosion film on the internal piping surfaces of the plant, in place of the radioactive cobalt-60. Cobalt-60 results from the neutron activation of stable cobalt that is found in various high wear reactor system components, as well as from the steel piping components. Corrosion and wear products from these sources are activated when passing through the neutron flux near or in the reactor core and then tend to plate out in the system internals and piping. Zinc injection has been used at Hope Creek Station essentially since plant startup, approximately seven years ago. Several interfering conditions have resulted in the eventual buildup of radioactive material into the corrosion layer of station piping, causing a radioactive source term similar to that at comparable nuclear power plants without zinc injection. One complication involved the high levels of iron that historically have been a problem in the feedwater system at Hope Creek. The iron readily compounds with the injected zinc, and has lowered the amount of free zinc available for piping plateout. The licensee estimated that as much as 85% of the injected zinc may have been lost to this effect. Another complication was that natural zinc injected was composed of roughly 50% zinc-64. Zinc-64 is subjected to neutron activation in the reactor and becomes the radioactive isotope zinc-65. Therefore, some of the plated-out zinc was itself contributing to the source term. Since January 1993, the licensee has been injecting depleted zinc, which contains approximately 2% of zinc-64. The existing zinc-65 in the piping corrosion film will decay with a 244-day half-life and in time may result in an overall decrease in piping source term.

By taking independent dose rate readings in the primary containment (drywell), in the torus and residual heat removal (RHR) system areas, and from a review of licensee survey maps, the inspector determined that the Hope Creek Station source term or radiation fields were typical of a non-zinc injected plant. The inspector noted that the licensee determined that the increasing radiation fields common for a new plant had leveled off during the last two years and radiation fields were now considered stable. Based on the inspector's experience, radiation fields were as high as a normal boiling water reactor after seven years of operation that had not experienced any chemistry or fuel failure abnormalities.

Location	General Dose rates(mrem/hr)
Drywell basement	80 near recirc loop, 15 between loops
Drywell entry level	10-80
Drywell main vessel penetration level	50, 80-150 near shielded penetration.
Drywell feedwater/corespray penetration level	30-400
Drywell main steam penetration level	15-50
Top of torus outside shell (RHR sytem piping)	5-90
Inside torus catwalk	2-6
Refueling bridge	5-8

The inspector reviewed the licensee's as low as is reasonably achievable (ALARA) organization to determine whether the exposure reduction efforts were commensurate with the above radiological conditions. The ALARA organization consisted of a senior supervisor, two radiation protection planners who work in the maintenance planning group to review maintenance work packages for initiating RWPs and ALARA reviews, and three RWP/ ALARA review writers that also provide labor for lead shield installation and removal. Shielding packages inside the drywell and in the area of the torus modification were not comprehensive, although some additional shielding efforts were made during the latter part of this inspection. Several selected high dose rate areas were shielded with a single layer of 1/4-inch lead blankets, which resulted in some dose rate reduction, however significant dose gradients remained in these areas. Other candidate shielding areas, such as the main drywell access ladder to the upper elevations, were missed until identified by the inspector. The ALARA program also included hot spot tracking, with some limited system flushing activities. The current ALARA program was somewhat limited in resources and in program development to accomplish significant exposure reduction. Continued improvement in this program area is warranted.

7.0 Exposure Control Implementation

The inspector reviewed the major outage work areas of the plant and made the following observations.

- The HP control points were well staffed with well coordinated activities. A good level of surveys and job coverage oversight was maintained. An area for enhancement involved the use of continuous air monitors at the drywell to quickly indicate deteriorating air quality emanating from this high maintenance area.

- The access control station efficiently handled the heavy throughput of workers under outage conditions with good oversight from HP supervision.

- A high level of air samples (approximately 90) were taken per day during the outage and generally showed very low airborne activity. This was indicative of well controlled maintenance outage activities.

- Good communication and rapport existed between the HP staff and the workers as evidenced through attendance at several ALARA briefings and control point briefings.

- The drywell and torus room were posted as high radiation areas at the entrances, when most of the work spaces constituted only radiation areas. Guidance provided in NRC Information Notice No. 84-82 considers this a poor practice. Many areas of dose gradient existed within these areas without any work area postings to alert the worker to higher and lower dose rate areas. The licensee agreed that posting practices could be improved. The inspector observed that the postings in the torus work area were significantly improved towards the latter part of this inspection, with the addition of yellow ALARA Caution signs and red ALARA Warning signs. The licensee stated that further enhancements in postings would be evaluated.

- Workers in two different work areas were observed waiting in higher dose rate areas than in nearby lower dose rate areas. The lack of dose rate postings in these areas may have attributed to this condition.

- The RWP procedure allows the use of closed circuit television monitoring to be used in place of constant coverage by health physics personnel. The inspector discussed with the licensee the need for two-way communciation in addition to the video monitor, which would allow HP to actively control the work being monitored remotely. The licensee will evaluate this finding and make appropriate program changes, where necessary.

- The RWP No. 94-92 covered safety relief valve removal and replacement in the drywell. The ALARA review indicated that new electric hoists had been purchased and were recommended to speed up the transport of these valves to and from the drywell. The inspector observed this work performed utilizing manual chainfalls. The ALARA staff was unaware that the electric hoist recommendation was not carried out. The inspector determined that insufficient ALARA implementation oversight was provided.

- There was some level of generic HP guidance written into many of the RWPs and the most common intermittent HP coverage frequency specified once per eight hours. The specificity of these controls should be enhanced.

8.0 Air Sampling Technical Basis

The inspector reviewed the derived air concentration (DAC) calculation method used by the counting laboratory. The inspector also reviewed the latest (June 17, 1993) offsite laboratory analysis of Hope Creek smear samples as given below. These samples represented a 10 CFR 61 waste stream analysis for station general contamination that was used to infer airborne radioactivity.

Isotope	Fraction	Radiation	DAC[1]	% of DAC
Fe-55	71.3%	electron capture	2E-6	13.2
Cr-51	11%	320 keV gamma	8E-6	0.5
Mn-54	8.2%	835 keV gamma	3E-7	10.1
Fe-59	5.2%	1292 keV gamma	2E-7	9.6
Zn-65	2.2%	1116 keV gamma	1E-7	8.1
Co-60	1.5%	1173 keV gamma 1332 keV gamma	1E-8	55.5
Co-58	0.32%	811 keV gamma	3E-7	0.4
Ag-110m	0.09%	658 keV gamma	4E-8	0.83
Sb-124	0.045%	603 keV gamma	1E-7	0.17
Pu-241	0.0012%	21 keV beta max.	3E-10	1.5

From the isotopes listed, 71.3% of the radioactivity was represented by non-gamma emitting isotopes, which were not measurable by the licensee's gamma spectroscopy analysis and were not included in the licensee's DAC reporting. The significance of this was low because the low energy beta emitting isotope (Pu-241) and Fe-55 represent only a small fraction of the DAC. The inspector determined that the licensee has not accounted for approximately 14.7% of the DAC value based on the above smear sample analysis data. It is an area for program improvement that the non-gamma emitting isotopes have not been considered in the DAC determinations. However, as previously discussed, the licensee has a good history of low activity in air samples and therefore, the additional DAC amount would not have resulted in any recordable internal exposures for the workers at Hope Creek Nuclear Generating Station. The licensee has agreed to review this issue and determine if any refinements to the program are necessary.

9.0 Exit Meeting

The inspector met with licensee representatives at the end of the inspection, on March 18, 1994. The inspector reported the inspection results and the licensee acknowledged the inspection findings.

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[1] As derived from the licensee's "Air Sampling Program Technical Basis", Revision 1.