U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-282/82-07(DETP); 50-306/82-07(DETP)

Docket No. 50-282; 50-306 Licenses No. DPR-42; DPR-60

Licensee: Northern States Power Company
414 Nicollet Mall
Minneapolis, MN 55401

Facility Name: Prairie Island Nuclear Generating Plant, Units 1 and 2

Inspection At: Prairie Island Site, Red Wing, MN

Inspection Conducted: May 5-7, 1982

L. J. Hueter

Inspector: L. J. Hueter 6/4/82

L R Greger

Approved By: L. R. Greger, Chief 6/4/82
Facilities Radiation
Protection Section

Inspection Summary

Inspection on May 5-7, 1982 (Reports No. 50-282/82-07(DETP); 50-306/82-07(DETP))
Areas Inspected: Special, announced inspection of actions taken in response to post-TMI requirements. The inspection involved 24 inspector-hours onsite by one NRC inspector.
Results: No items of noncompliance or deviations were identified.

DETAILS

1. Persons Contacted

*A. Johnson, Radiation Protection Supervisor
J. Lemmerman, Radiation Protection Specialist
J. Maurer, Radiation Protection Specialist
*F. Tierney, Jr., Plant Manager
*D. Schuelke, Superintendent, Radiation Protection
*E. Watzl, Plant Superintendent, Engineering and Radiation Protection

B. Burgess, NRC Resident Inspector
*C. Feierabend, NRC Senior Resident Inspector

*Denotes those attending the exit interview.

2. General

This inspection, which began about 11:30 a.m. on May 5, 1982, was conducted to review the status of licensee actions regarding post-TMI requirements. Tours included the reactor control room; the Technical Support Center in the administration building, the hot sample room on the 715' elevation of the auxiliary building; the main steam line noble gas effluent monitor installations located on various levels of the auxiliary building; and the liquid sample dilution room ("Hot Cell") located in the lower level of the turbine building.

3. Status of NUREG-0737 Items

During a previous inspection[1], the status was determined for several NUREG-0737 items (long term). At that time, several of the items were not completed and/or were not inspected. An update on the status of these items is presented below. No deviations were identified from the status of these items as reported in the licensee's April 16, 1982 letter from L. O. Mayer in response to D. F. Eisenhut's Generic Letter 82-05.

Although no Technical Specification changes have been made regarding the NUREG-0737 modifications for coping with post accident conditions, the licensee plans to submit appropriate changes when specifically requested by the NRC. The licensee stated that recent correspondence indicates that a request will be forthcoming.

a. Post-Accident Sampling and Analysis Capability (Task II.B.3.2.B)

Although the licensee considered the post-accident sampling and analysis system "operational" by the NUREG-0737 specified implementation date of January 1, 1982, in their letter dated

[1] Inspection Report No. 50-282/82-03.

December 31, 1981, the licensee stated that sample valves on loop B of Unit 2 had not been replaced with fully qualified valves. Further, this letter noted that the replacement could not be completed during the last Unit 2 refueling because of equipment delivery problems and requested an extension of the implementation date to the end of the 1982 refueling outage (summer 1982). This request for extension was repeated in the licensee's April 16, 1982 letter. The April 16 letter also noted that as an interim measure samples could be obtained from the RHR sample line in the event of failure of the loop B sample valves.

Both the reactor coolant and the containment atmosphere accident sampling systems are located in the Hot Sample Room, which is also used for routine reactor coolant sampling. This room is located on the 715' elevation of the auxiliary building and is adjacent to the Hot Chemistry Laboratory and a short distance from Unit 1 and Unit 2 containments. To provide for accident sampling, modifications such as adding or altering sample, drain, purge, and flush lines and adding shield walls and encasing piping in lead shielding have been made. Drag valves in the reactor coolant sample lines are designed to provide flow restriction to limit reactor coolant loss in the event of sample line rupture. The ventilation exhaust from the sampling hood is filtered through HEPA filters and charcoal adsorbers. Drain lines are diverted to sumps that pump back to containment under accident conditions. Tools for remote handling and inner and outer lead pigs for sample transport were in place.

An area monitor is located in the sample room with the meter located just outside the entrance to the room. Breathing air (air line) is available at the entrance to the sample room as well as to the Chemistry Laboratory. SCBA can be used to reach the area; then the air supply can be changed to the air line while in the area and back to SCBA for exit from the area.

According to licensee personnel, neither reactor coolant or containment atmosphere sampling require an isolated auxiliary system to be placed back in service in order to utilize the sampling system.

The licensee has demonstrated that with primary coolant activity of 10 Ci/ml a 10-20 ml sample can be collected in an 8-10 R/hr field in a time of about 20 seconds resulting in a body dose of about 55 mrem with little additional dose to the extremities due to use of handling tools. The above time and exposure estimates include valve operations to purge sample lines. The sample is placed in a small lead pig at the sample location which in turn is placed in a larger lead pig on wheels located outside the sample room. The sample is transferred to the "Hot Cell" room located on the 695' elevation (ground floor) of the turbine building. This room has a filtered hood for use in sample splitting, dilution, and analyses. Survey meters, a frisker, an air sampler

(CAM), protective clothing, lead shielding, emergency procedures, and other equipment and supplies for emergency use are located in the room.

For chloride analysis, 1 ml of reactor coolant sample is diluted with 100 ml of demineralized water. The diluted sample is then used in a turbility test using a Hach Turbidimeter having a chloride sensitivity of 0.1 ppm. The licensee has demonstrated that the reactor coolant sample can be collected and the chloride analysis performed onsite in about an hour, well within the NUREG-737 specified time (three hours).

For boron and radiological analyses, 1 ml sample of reactor coolant sample is diluted with 1000 ml of demineralized water. Five ml of the diluted sample is used in the boron test, which uses the carminic acid technique and the associated color belt with a boron sensitivity of about 1 ppm in the diluted sample (corresponding to 1000 ppm in the primary coolant). The sampling and analysis for boron can be completed in about two hours.

Ten ml of the diluted reactor coolant sample (further dilution of the sample is provided if the sample has a radiation level exceeding 3 mR/hr at contact) is used for radionuclide analysis in the emergency Counting Trailer currently located near Unit 1 containment. The trailer houses an high efficiency germanium detector, with provisions to protect the detector in the event of low liquid nitrogen level, and a multichannel analyzer which has a library of radionuclides that are indicators of the degree of core damage. A weekly efficiency check is conducted to assure the unit is maintained in a proper working status. The sampling and radioanalysis can be accomplished in about one hour. The licensee has plans to replace the emergency counting facility housed in the trailer with a GeLi counting facility to be located in the training building about three quarters of a mile west of the reactor facility.

The licensee evaluation shows that the total exposure received while performing the reactor coolant sample dilutions in preparation for chloride, boron, and radionuclide analyses is about 250 mrem total body and about 1 rem extremities. The sample collection and sample dilution would most likely be performed by different individuals.

To determine the hydrogen, oxygen, and noble gas concentrations in the primary coolant during accident conditions, a "Beckman Panel" system is provided in the Hot Sample Room. The system includes: (1) a 75 cc sample bomb in which a known volume of primary coolant is introduced at about 50 psi; (2) a 150 cc evacuated bomb for expansion of the coolant sample; (3) a hydrogen and oxygen analyzer; (4) a pump for circulating nitrogen through the sample and the rest of the system; (5) a gas sample vial for collecting a gas sample (following equilibrium conditions as indicated by the hydrogen and oxygen analyzer) for analysis of the noble gas activity, and (6) associated piping and valving for introduction of primary coolent and a nitrogen purge and for isolating various components. Licensee tests indicate nearly 100 percent stripping factor for the entrained gases. The hydrogen and oxygen analyzer is calibrated weekly for hydrogen with one percent hydrogen and monthly for hydrogen and oxygen with five percent hydrogen, 50 percent hydrogen and 21 percent oxygen (atmospheric air).

Containment atmosphere sample lines pass through a small condenser from which a sample of condensed steam can be collected and analyzed if desired. Next, the gas sample, if low in activity, can be directed through a removable particulate filter and silver zeolite adsorber (for use in determining concentrations of particulates and iodines) before collecting several ml of gas to be analyzed for noble gas concentration.

For higher activity atmospheres, provision is made to collect a small unfiltered sample which can be analyzed for particulate, iodine, and noble gas activities simultaneously. In either case, a filtered gas sample is passed through a gas analyzer which analyzes for both hydrogen and oxygen concentration. The readout of this analyzer is located in the Hot Chemistry Laboratory. Waste gas, including the nitrogen purge from this sample system, goes to the vent header where it is compressed and stored.

While the licensee has compared the xenon concentration in containment as determined from the post-accident sampling system with that determined by the normal sampling system, no such comparison has been made since some of the more recent modifications were made to the post-accident sampling system. This matter was discussed at the exit meeting.

Written procedures for conducting post-accident sampling, and training of radiation/chemistry personnel in the procedures, has been accomplished. During a tour, the individual accompanying the inspector demonstrated a good knowledge of the procedures.

In conclusion, while the post-accident sampling capability is operational, loop B sample valves for Unit 2 must still be replaced with fully qualified valves. As the valve replacements should be done with the plant in a cold shutdown condition, the licensee currently plans to complete this replacement by the end of the 1982 refueling outage, which is scheduled to begin in June. The licensee notes that in the event of failure of the loop B sample valves in Unit 2, samples may be obtained from the RHR sample lines.

It appears the licensee has met the intent of long term NUREG-0737, Item II.B 3.2.B. Replacement of the Unt 2 loop B sample valves will be reviewed during a future inspection. (306/82-07-01)

b. Extended Range Noble Gas Effluent Monitor (Task II.F.1.1.B.2)

Two potential release paths have been identified as requiring extended range noble gas monitoring. These paths are the shield building vent for each unit and the main steam line for each unit. During a previous inspection[2], it was noted that the licensee had implemented the NUREG-0737 criteria for extended range noble gas monitoring for the shield building vent pathways but not for the main steam line pathway. The delay for the latter was due to a faulty electrical component. As a result, the licensee, by letter dated December 31, 1981, requested a delay for the implementation date to March 1, 1982, for the main steam line monitors. Licensee letter of March 1, 1982, requested a further extension of the implementation date to April 30, 1982. Replacements for faulty components were received and installed to permit completion of testing, calibration, and implementation by April 30.

A section of each of the four main steam lines (Loop A and Loop B of each reactor unit) in the auxiliary building is "viewed" through a collimated lead shield (6 to 10 inches of lead) by a very small Victoreen G-M type exposure rate meter.

All four shielded detectors were observed by the inspector during a tour. Monitor readings in mR/hr over the range of 1E0 to 1E5 are provided in electrical buss room number 12, located in the turbine building, and in the technical support center. Operable readouts with recording capabilities were observed by the inspector during tours of both locations. When the recorders are not in use, as is the case during normal operation, the technical support center unit has the ability to recall the data from the past hour. Emergency power is provided to the detector, monitor, and computer system.

During the tour, it was observed that two of the four monitors in electrical buss room number 12 were not reading high enough on the low end of the scale to prevent the monitor from indicating a down scale trip, which implies an inoperable monitor. The monitors were observed, however, to be operable. This condition was discussed at the exit meeting.

Curves have been generated to relate exposure rate readings to noble gas concentrations in the steam. These curves are compensated for changing isotopic ratio with length of time since shutdown and intervening shielding of the steam line and insulation. This data, along with data from instruments providing various valve positions and flow rates, is fed directly into a computer program to provide live time release rate as well as integrated release.

[2] Ibid.

The range of the instruments is more than adequate to provide the NUREG-0737 specified design basis concentration of 1E3 μCi/cc xenon-133 equivalent (corresponding to about 3E4 mR/hr).

Each of the four main steam line noble gas effluent monitors was calibrated in April 1982. Calibration at each decade over the full range of the monitor was performed in the calibration facility, followed by calibration on the two lower decades after installation. The licensee plans to perform a monthly functional test, and, at refueling intervals a calibration check.

It appears the licensee has met the intent of the long term NUREG-0737 Item concerning extended range noble gas effluent monitoring for all identified potential release paths.

c. Containment High Range Radiation Monitor (Task II.F.1.3)

The inspector reviewed the location of the two high range radiation monitors (General Atomic) in each containment. Operable readouts with recording capabilities were observed during tours of the reactor control room and the technical support center. When the recorder is not in use, as is the case during normal operation, the technical support center unit has the ability to recall the data from the previous hour.

Environmentally qualified cable was received in time to meet the extended implementation date of March 1, 1982. The inspector reviewed Work Request Authorization (WRA) packages FO 513-RD-Q and FO 514-RD-Q for routing and terminating cables per design change installation procedure. The packages, including sign-offs and stampings indicating QC inspection and QC acceptance, appeared to be complete.

Each detector consists of a single ion chamber, with a thin stainless steel wall designed to measure gamma radiation over the range of 1E0 to 1E8 R/hr. Logarithmic readout is provided.

Vendor supplied data of special environmental qualifications of the equipment was reviewed. The data demonstrated an energy range response as specified in NUREG-0737 (60 keV to 3 MeV) with essentially a flat response for gamma energies over this energy range. Linearity of response to source intensity was demonstrated by a 13 point check ranging from 6.5E-1 R/hr to 5.0E6 R/hr.

Calibration data showed all four units were initially calibrated during January and February 1982. All four detectors were calibrated in the licensee's calibration facility using a 20 Ci cesium-137 source at 5 R/hr and 10 R/hr. In addition, a reading of about 50 R/hr was obtained with the source in contact with the detector. Following this, in situ calibration by electronic signal substitution was performed over the range of 1E0 to 1E7 R/hr and system response was checked using both a 143 mCi cobalt-60 source and an 11 mCi cesium-137 source.

A monthly functional test, which involves an electrical signal input corresponding to 1E5 R/hr, has been initiated. The licensee plans to perform electronic calibrations and source response checks at refueling intervals.

It appears the licensee has met the intent of the long term NUREG-0737 Item II.F.1.3.

4. Increase in Unit 1 Coolant Activity Starting in November 1981

The inspector reviewed radiation protection aspects of the increase in reactor coolant activity in Unit 1 which began in November 1981 shortly after a refueling outage. While only a small increase in noble gas concentration occurred in the reactor coolant, the iodine-131 concentration increased from normal levels of less than 0.001 uCi/ml to a peak of 0.460 uCi/ml during the period of November 17-18, 1981. The fuel vendor was promptly consulted and after reviewing data estimated that probable cladding failure had occurred on about 15 fuel pins. A reduction in power to 90 percent on the evening of November 18, resulted in an immediate decline that basically followed the pattern predicted by the fuel vendor. Within a few days, an equilibrium condition was reached. The iodine-131 concentration leveled out at about 0.05 μCi/ml.

The inspector's review of the matter identified no increase in plant radiation levels except in the immediate area of the VCT (due to xenon gas). No iodine was detected in liquid releases or in radwaste packages shipped to burial. While a small increase in iodine-131 activity in gaseous releases was identified during the fourth quarter of 1981, it remained below one percent of the design objective. The small increase in noble gas activity was too low to be detected by the effluent monitor but was detected and quantified by routine grab sample analysis.

At the next refueling outage, the licensee plans to remove the new fuel put in the reactor at the last refueling outage. Fuel sipping will be conducted on this fuel and identified leakers replaced.

No items of noncompliance or deviations were identified.

Exit Meeting

The inspector met with licensee representatives (denoted in Section 1) on May 7, 1982. The following items were discussed:

a. The purpose and scope of the inspection.

b. The licensee agreed to compare noble gas concentration in containment as determined by the post-accident sampling system with that determined by the normal sampling system to assure that a proper sample can be obtained through the post-accident sampling system. (Section 3.a)

c. The inspector noted that although two of the four extended range noble gas effluent monitors (for the main steam lines) were operating, they read so low that down scale trip indicators implied the units were inoperable. Licensee personnel stated that the two original detectors in question were defective and that the replacement detectors obtained did not contain small internal sources needed to keep the monitor on scale. It was stated that new detectors with the internal sources are being ordered. (Section 3.b)