U. S. ATOMIC ENERGY COMMISSION
DIVISION OF COMPLIANCE

REGION I

CO Inspection Report No. 50-146/71-03

Subject: Saxton Nuclear Experimental Corporation

License No. DPR-4

Location: Saxton, Pennsylvania

Priority

Category C

Type of Licensee: W 28 MWt, PWR

Type of Inspection: Routine, Announced

Dates of Inspection: December 27 - 29, 1971

Dates of Previous Inspection: September 7 - 9, 1971

Principal Inspector: R. L. Spessard 2/9/72
R. L. Spessard, Reactor Inspector — Date

Accompanying Inspectors: None — Date

Date

Other Accompanying Personnel: None

Reviewed By: G. L. Madsen 2/9/72
G. L. Madsen, Reactor Inspector — Date

Proprietary Information: None

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## Section I

Enforcement Action:

A. Technical Specification I.8.d - Accidental releases of gaseous activity from the plant stack in excess of specified limit. (Paragraph 12)

B. 10 CFR 50.59 - System modification without a written safety evaluation. (Paragraph 12)

C. Technical Specification N.2.b.(2) - Failure of operating personnel to follow a written emergency procedure. (Paragraph 12)

Licensee Action on Previously Identified Enforcement Matters:

None required.

Unresolved Items:

The primary coolant had not been sampled for impurities since returning to operation. (Paragraph 13)

Status of Previously Reported Unresolved Items:

A. Auxiliary systems modifications to permit increasing the primary coolant hydrogen concentration have been completed and appropriate facility procedures have been reviewed. However, additional records are required to complete the QA package. (Paragraph 14)

B. Facility test procedures have been revised to include acceptance limits or criteria. This item is considered closed.

C. A change report describing modifications made to the RWDF evaporator feed system was submitted to DRL by letter dated December 9, 1971. This item is considered closed.

Unusual Occurrences: None

Persons Contacted:

C. R. Montgomery, President, SNEC
D. A. Goodman, Supervisor, Operations and Testing
W. E. Potts, Supervisor, Reactor Plant Services
R. E. Beale, Radiation Protection Engineer
G. Reid, Radiochemist
R. Walton, Radiochemist
E. Hooper, W On-Site Engineer

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T. Osborn, Shift Supervisor (SRO)
F. Rodies, Operator (RO)
P. Trexler, Operator (RO)
F. Hertrich, Operator (RO)

Management Interview:

The following subjects were discussed with Mr. Montgomery, President, SNEC, on December 29, 1971:

A. Accidental Releases of Gaseous Activity

The inspector stated that the modification made to the caustic addition line to prevent a recurrence of stem packing leakage on valve (V-1231) was made without a written safety evaluation as required by 10 CFR 50.59. After further discussions on 10 CFR 50.59 and the importance of written safety evaluations, Mr. Montgomery stated that a written safety evaluation would be prepared and the modification would be documented in Saxton's Monthly Report. Mr. Montgomery further stated that the requirements of B31.1.0 - 1967 Edition would be met for this modification.

The inspector stated that a review of the events pertaining to these releases disclosed that operator actions specified in Procedure EI-510 had apparently not been adhered to. Mr. Montgomery stated that when reviewing the events of the December 15, 1971 release, it should be obvious to anyone that the emergency instruction was not followed. He further stated that following the November 29, 1971 release, EI-510 was placed in the control room and all licensed SROs and ROs were required to read and initial the procedure. Mr. Montgomery stated that following the December 15, 1971 release, he had personal discussions with SNEC licensed personnel on this matter. He further stated that the complete details of these releases would be reviewed by the SNEC Safety Committee during their next meeting.

The inspector indicated that enforcement action would probably be taken on the matters discussed above. (Paragraph 12)

B. Auxiliary Systems Modifications

The inspector stated that records indicated an independent audit of the QA package had been performed by GPU and MPR Associates; that facility procedures affected by this modification had been revised; and that these items were considered resolved.

The need for additional records pertaining to three welds in order to complete the QA package was discussed. Mr. Montgomery stated that Westinghouse would be contacted on this matter, and that the information would be added to the QA package. Mr. Montgomery was informed that this matter was considered to be unresolved and that these records would be reviewed during the next CO inspection. (Paragraph 14)

The inspector indicated that the welder's qualification records for procedures used to weld austenitic corrosion resisting steel pipe and tubing disclosed that the welders were qualified to the procedures in accordance with ASME Boiler and Pressure Vessel Code, Section IX, but that the welders had not been qualified to the procedure until after the field welding was completed. Mr. Montgomery stated that following the audit performed by GPU and MPR, the qualifications of the welders at the time the field welding was performed were questionable, and therefore, it was decided to qualify the welders to the procedures used to insure that qualified welds could be made. Mr. Montgomery was informed that this matter would receive further review at CO:I.

In a subsequent telephone conversation on January 21, 1971, Mr. Swift, Nuclear Plant Superintendent, was informed that the welder qualifications for these procedures had been reviewed in CO:I and were found to be acceptable.

C. Facility Test Procedures

The inspector stated that records indicated testing procedures had been revised to include acceptance limits or criteria; that the revised procedures were being used; and that this matter was considered resolved.

D. Change Report No. 27 - RWDF Modification

The inspector stated that the change report had been submitted to DRL and that this matter was considered resolved.

E. Sampling of Primary Coolant for Impurities

The inspector stated that chemistry records indicated that the primary coolant had not been sampled for impurities since resumption of operations. Mr. Montgomery stated that based on plant experience and the midlife fuel examination results for crud buildup, he was certain the Technical Specifications limit was not being exceeded. He stated that an analysis would be performed in the near future. The inspector stated this matter was considered to be unresolved and would be reviewed during the next CO inspection. (Paragraph 13)

F. Core III Midlife Core Physics Test Results

The inspector stated that all of the core physics test results had not been supplied to SNEC by Westinghouse and, therefore, the inspector's review of this matter was not complete. Mr. Montgomery stated that because of the holidays, Westinghouse's review of the data had not been completed. He stated that the data should be completed in about another week and that the data would be communicated to the inspector by telephone.*

G. Containment Air Particulate Monitors

The importance of these monitors and the condition whereby they have been frequently out of service for extended periods of time were discussed. Mr. Montgomery stated that this matter would be reviewed with Mr. Swift, and that corrective action would be taken to minimize this condition. The inspector stated this matter would be re-inspected during the next CO inspection. (Paragraph 15)

*Data provided in a subsequent telephone conversation on January 14, 1972.

## Section II

**Additional Subjects Inspected, Not Identified in Section I, Where No Deficiencies or Unresolved Items were Found**

1. General

Since the last CO inspection, the licensee completed the following significant tasks: the auxiliary systems modifications to permit increasing the primary coolant hydrogen concentration, reactor vessel loading, control rod testing, a four hour hot hydrostatic leak test of the primary coolant system (in service inspection), zero power physics testing and startup training (November 10 - 17, 1971), power physics testing (November 18 - December 6, 1971), AEC license examinations for two operators (December 7 and 8, 1971), and full power operations and load cycling (December 9 - 25, 1971).

At the time of this inspection, the reactor was in a hot critical condition following an unscheduled scram that occurred on December 25, 1971. The inspector observed reactor startup from this condition, load cycle operations, a demonstration of an in-core flux map, charging and letdown operations, RIC-3 response during sampling of the primary coolant, the operation of the stack "Cal" damper, and the surveillance test of the safety injection and recirculation systems pumps.

There were two unscheduled scrams following completion of core physics testing which resulted from equipment malfunctions.

2. Administration and Organization

   a. Personnel changes
   b. On-Site Safety Committee meeting minutes, September 10 through December 29, 1971
   c. SNEC Safety Committee meeting minutes, August 31 through December 29, 1971

3. Operations

   a. Future facility plans
   b. Control Room Log Book, October through December 29, 1971
   c. Monthly Operating Reports, October through December 29, 1971
   d. Operator performance (observations during the inspection)

4. Maintenance Records

   a. CRDMs and scram breakers (W, Model DB-15)

b. 480 V MCC breakers
c. Underground storage tanks - cathodic protection system
d. RWDF preventive maintenance - valves
e. Experience with limitorque operators
f. Inspection results of bottom core plate and pilot tubing and upper core barrel and support plate

5. Facility Procedures

a. The following procedures were reviewed:

(1) OI-418 Boric Acid Removal or Dilution
(2) OI-454 Crud Filter System
(3) EI-510 High Radioactivity Level
(4) SO-8 Level and Pressure Limits for Purification Surge Tank
(5) SO-16 Minimum Boron Concentration Requirements

b. The following facility test procedures have been revised to include acceptance limits or criteria:

(1) MI 622 Source calibration of the steam generator blowdown radiation monitor (RIC-5)
(2) MI 623 Source calibration of stack effluent radiation monitor (RIC-3)
(3) MI 624 24 hour leak test of underground liquid storage tanks
(4) MI 625 No. 2 turbine overspeed trip test
(5) MI 621 Control rod drop time test
(6) MI 626 Level test of underground liquid storage tank annuli
(7) MI 5 Reactor plant alternate power supply automatic transfer test
(8) OI 413 Nuclear instrumentation
(9) OI 414 Radiation monitoring system
(10) MI 610 Safety injection system test
(11) MI 617 Scram circuit response time

c. The following facility procedures have been revised to incorporate the changes involved in the reactor auxiliary systems modifications:

(1) OI 402a Normal reactor startup from hot shutdown
(2) OI 403a Reactor startup following scram
(3) OI 407 Main coolant system cooldown
(4) OI 408 Filling and venting of the main coolant system
(5) OI 419 Corrosion control agent preparation and addition
(6) OI 423 Shutdown cooling
(7) OI 438 Instrument and plant air supply, containment vessel
(8) OI 440 Shutdown from power operation
(9) OI 417 Boric acid addition to main coolant

(10) OI 418 Boric acid removal or dilution
(11) EI 509 Loss of main coolant
(12) EI 510 High radioactivity level
(13) EI 511 Malfunction of pressurizer power operated re...ef and safety valves
(14) EI 513 Failure of regenitive heat exchanger
(15) EI 515 Malfunction of let down flow control
(16) EI 516 Loss of component cooling

Note: The revisions to the procedures (b and c above) were reviewed by the inspector.

6. Facility Surveillance Test Requirements

The inspector reviewed the following completed test procedures and other log sheets for the period September 10 through December 29, 1971:

a. Weekly Tests

(1) Primary versus secondary heat balance
(2) Core flux distribution
(3) Core reactivity
(4) Level tests of the underground liquid storage tank annuli (3 times/week)

b. Monthly Tests

(1) Calibration of the failed fuel element detector
(2) Calibration of the steam generator blowdown radiation monitor
(3) Calibration of the stack effluent radiation monitor
(4) Calibration of the radiation monitoring system
(5) Safety injection and recirculation pumps and automatic startup control (observed the December 29, 1971 test)

c. Semi-annual Tests

(1) Control rod drop time
(2) Scram circuit response
(3) No. 2 turbine overspeed trip
(4) 24 hour leak test of underground liquid storage tanks

7. Primary System

a. Primary coolant chemistry logs (November - December 1971) for the following parameters which have Technical Specification limits:

(1) Chlorides

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(2) Oxygen
(3) Boron
(4) Lithium
(5) Hydrogen
(6) Radioactivity

b. Make-up water to reactor plant chemistry logs (November - December 1971) for the following parameters which have Technical Specification limits:

(1) Conductivity
(2) Chlorides
(3) Silicon dioxide

c. Hydrogen addition control, procedures and records (November - December 1971)

d. Four hour hot hydrostatic test (in service inspection), completed test procedure, dated November 8, 1971.

8. Reactivity Control and Core Physics

a. 1% shutdown margin requirements

b. Core power distribution limits (data from December 13, 1971 flux map) for the following:

(1) Loose lattice assemblies, peak pellet
(2) Load follow assemblies, peak pellet
(3) PNC assembly at peripheral core position D-5, peak pellet
(4) Test fuel rods for load follow assemblies (TS Change 44), peak pellet
(5) Loose lattice rods surrounding solid zirc-4 rod (TS Change 47), peak pellet
(6) PNC test fuel rods for loose lattice assemblies (TS Change 41), peak pellet
(7) Power transient test fuel rods for peripheral subassembly at core position N-3 (TS Change 48), peak pellet
(8) Plutonium oxide fuel rods for central subassembly (core position N-1), peak pellet

c. Midlife core physics test results for the following parameters:

| Core Parameter | Core Power |
|---|---|
| (1) Mod. Temp. Coeff. | |
| ARO (All rods out) | HZP (Hot Zero Power) |

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| | |
|---|---|
| R2 In (Control rod No. 2) | HZP |
| ARO | 20 MW |
| R2 In | 23 MW |
| (2) Power Coeff. | 10-20 MW<br>20-25* MW |
| (3) Boron Worth | HZP<br>20 MW |
| (4) Peak Xenon Worth | 23 MW<br>6 Hrs.<br>after shutdown |
| (5) Pressure Coeff. | HZP |
| (6) (a) Rod 2 Worth | HZP<br>10 MWt |
| (b) Rod 5 Worth | HZP |

9. Auxiliary Systems

Regenerative heat exchanger - the QA package for the relocation of the safety valve, V-53, was reviewed to insure that the safety considerations described in the licensee's Change Report No. 26, dated December 9, 1971, had been met.

10. Containment

Hydrostatic test records of two containment vessel (CV) penetrations used for the pipe that connects PSV-501 (purification surge tank relief valve) to the CV discharge tank (October 28, 1971). Note: This line was rerouted and therefore both penetrations were required to be tested by the Technical Specifications.

11. Radiation Protection

a. Radioactivity sampling records of the refueling water storage tank, September through December 1971
b. Personnel exposure records, August through October 1971
c. Observation of a radiation survey of the primary system letdown line performed by one of the shift operators
d. Gaseous and liquid release permits, September through November 1971
e. Control of gaseous releases during primary coolant sampling and operation of the charging pumps (required because of high gaseous activity, mostly xenons, in the primary coolant)

*Extrapolated from 23 MWt; 25 MWt projected 100% power.

Details of Subjects Discussed in Section I

12. Accidental Releases of Gaseous Activity from the Plant Stack

References: Inquiry Report Nos. 50-146/71-02 and 03 and Licensee letters to DRL, dated December 9 & 28, 1971

Additional information relating to these releases which was obtained during discussions with Messrs. Montgomery and Potts and from a review of site records, procedures and equipment is summarized as follows:

a. November 29, 1971 releases - There were three separate releases from the same source (stem packing leakage from valve V-1231 in a caustic addition line which is connected to the purification system letdown piping in the charging pump room) during a seven hour period. The release concentrations, excluding I-131 and averaged over a 15 minute period, for these releases were $3.52 \times 10^{-3}$, $0.81 \times 10^{-3}$ and 1.08 x 10-3 uCi/cc respectively. The release concentrations, averaged over a 15 minute period, for the first and third releases were in excess of the Technical Specification limit of $1 \times 10^{-3}$ uCi/cc. (Paragraph N.7.b)

Paragraph N.2.b.(2) of the Technical Specifications specifies, in part, that Standing Orders to operating personnel shall require that written procedures and instructions provided for emergency conditions shall be followed in conducting activities identified therein. SO No. 18 is used for promulgating these requirements. Emergency instruction for High Radioactivity Level (EI-510) requires, in part, that when the stack gas (RIC-3) alarm is actuated, immediate action consisting of the following will be taken:

(1) Attempt to reset the alarm and check the instrument for proper operation and verify the alarm with a portable monitor or air sample.

(2) Temporarily place the fresh air damper to the "Cal" position. If the counts do not decrease, this is evidence of increased background. If the counts do decrease, this is evidence of a radioactivity release and the following action is required.

(3) Terminate or reduce the rate of activity release resulting from the following:....(6) Boron dilution....(9) Pressurizer vent line operation.

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During the first release and after RIC-3 had been verified to be operating correctly (by an instrument technician), a period of about 10 minutes elapsed before action was taken to reduce the rate of activity release, i.e., boron dilution terminated by operator action from the control room. RIC-3 indicated a release concentration of greater than $1 \times 10^{-3}$ uCi/cc for a period of about three minutes prior to isolation and a concentration of $1.17 \times 10^{-3}$ uCi/cc at the time of isolation.

The ten minute period which passed before action was taken to terminate the release does not appear to meet the immediate action requirements specified in EI-510.

The licensee's investigation to identify the source of leakage and the corrective actions taken (tightening of valve packing on suspected valves in the charging pump room and operating alternate charging pumps for charging and letdown) prior to the third release indicated that the source of leakage was some where in the bleed line downstream of valve (V-114). The third release occurred when bleed and feed operations were resumed (V-114 opened). The source of leakage was identified on the following day be pressure testing the bleed line (from valve V-114 to the RWDF storage tank) with nitrogen at 100 psig, and the leakage was stopped by tightening the packing nut on valve V-1231. A review of the events disclosed that the packing nuts on valves V-1231 and V-1237 were tightened after the second release and not after the first release as previously reported by Mr. Montgomery during a telephone conversation on December 1, 1971 (CO Inquiry Report No. 50-146/71-02).

The inspector asked Mr. Montgomery why bleed and feed operations were resumed following the second release since the source of leakage had been isolated to this line and equipment for pressure testing this line with nitrogen was readily available for use. Mr. Montgomery stated that bleed and feed operations were resumed to see if the source leakage in the line could be identified and to check whether or not the corrective actions taken (tightening of valve packing on valves connected to this line) had been effective. The inspector stated that it would seem more prudent to perform this check with nitrogen rather than with xenon gases.

The corrective action taken by the licensee consisted of the following:

(1) Operate with globe valve V-1231 open and back seated so the stem packing is not pressurized by the letdown flow.

(2) A second globe valve, V-2461, was installed in the caustic addition line outside of V-1231 to provide isolation capabilities.

(3) The packing of all valves and all swagelock fittings in the charging pump room have been tightened. The stem packings and bonnet gaskets of about 25 valves in the underground storage tank piping have been leak tested and the remaining valves in these systems are scheduled to be leak tested as operations permit.

(4) The licensee's letter to DRL, dated December 9, 1971 with a cover letter of the corrective actions taken was sent to all members of the SNEC Safety Committee.

When the inspector asked Mr. Montgomery to show him the written safety evaluation covering the installation of valve V-2461, Mr. Montgomery stated that a written evaluation had not been prepared. He stated that this change had been discussed among supervisory personnel and it was concluded that this change would be a prudent one.

b. December 15, 1971 release - Moments after the vent line was put in service, RIC-3 began to increase and within five minutes its alarm point was reached. Then, a period of about 21 minutes elapsed before the fresh air damper was placed to the "Cal" position to verify the release and about one minute later the vent line was isolated by operator action from the control room. RIC-3 indicated a release concentration of greater than $1 \times 10^{-3}$ uCi/cc for a period of about 14 minutes prior to operation of the fresh air damper and a concentration of approximately $1.22 \times 10^{-3}$ uCi/cc when the damper was operated. During the release and prior to operation of the fresh air damper, radiation surveys were made of the plant including charging room, auxiliary equipment room, yard area (includes RIC-3) and RWDF. The surveys revealed an increase in background activity; however, Saxton personnel did not interpret the survey results to be indicative of a release.

The 22 minute period which passed before action was taken to verify the release, as indicated by RIC-3, and reduce the rate of activity release does not appear to meet the immediate action requirements specified in EI-510.

The pressurizer vent line isolation valve, PC-97V, is a Mason Neilan, forged stainless steel 3/8" globe valve, rated at 2485 psig at 650°F. The valve packing (source of leakage) is John Crane, Type 2CR-J (asbestos graphite) with 10-11 packing rings and is recommended for liquid and gas service.

The pressurizer was being vented for the first time since returning to power on . cember 9, 1971. Prior to December 9, it had been vented intermittently for about 20 hours without incident. Mr. Montgomery stated that this operating period (venting prior to

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the release) had been mistakenly reported in SNEC's December 28, 1971 letter as having occurred during the period December 8-15.

Mr. Montgomery stated that the valve manufacturer had been contacted, and it was recommended that the valve be cycled several times and its stem packing tightened following each cycle until the packing was tight. He further stated that a hydrostatic test procedure to insure leak tightness of the stem packing would be prepared and reviewed by the safety committees prior to use. According to Mr. Montgomery, a pressure test with nitrogen up to 2000 psig would be possible. The vent line will remain isolated until PC-97 is satisfactorily tested.

13. Sampling of Primary Coolant for Impurities

The Technical Specifications (paragraph N.4.b.(6)) specifies that for power operation above 1 MWt, impurities in the primary coolant will be less than 5 ppm. During a review of chemistry records, the inspector noted that the primary coolant had not been sampled for impurities since returning to operations following the recent extended outage.

In separate discussions with Messrs. Reed and Potts, the inspector was informed that with the high gaseous radioactivity levels in the primary coolant, the Plant Superintendent, Mr. Swift, would not allow this sample to be taken using the former procedure because of the large volumes involved. The maximum valve (impurities) measured in the past was reported to be about 0.2 ppm. The crud buildup of fuel rods found during the midlife fuel inspection were reported by Mr. Potts to be minimal. Mr. Potts stated that there were plans to pull a sample in the near future using the newly installed crud sampling system.

14. QA Package - Auxiliary Systems Modifications

References: Licensee's Change Report No. 25, dated October 18, 1971 and Amendment No. 1, dated November 10, 1971, submitted to DRL

A review of QA records and discussions with Messrs. Goodman, Potts and Montgomery disclosed that the safety considerations described in paragraphs 3.1 through 3.6 of Change Report No. 25 had been met; however, the inspector identified three welds which required additional supporting records to complete the QA package. These were:

a. Shop weld on valve V-2456 (pressurizer vent line) - The only documentation available was a referenced Westinghouse job order number in the bill of materials.

b. Weld Nos. 44 and 51 (boric acid system) were identified on the weld inspection record as being repaired. The weld inspection record also showed that a satisfactory LP test of these welds had been performed. The welding procedure specified that weld defects and method of repair would be documented.

In a subsequent telephone conversation with Mr. Swift, Nuclear Plant Superintendent on January 21, 1972, the inspector was informed that the additional records for these welds were available and would be added to the QA package.

15. Containment Air Particulate Monitors

The containment vessel air is monitored for particulate activity by RIC 1 and 11 and for gaseous activity by RIC-2. The licensee's letter to DRL, dated October 5, 1971, describes methods for identifying leakage from the primary coolant system which includes detection by these monitors. The licensee considers, as stated in the letter, the particulate monitors to be more sensitive in leak detection than the gas monitor.

A review of equipment trouble reports for the period of September through December 28, 1971 revealed that RIC 1 and 11 experienced filter feed failures on a weekly basis during this period. In discussing this matter with Mr Potts and reactor operator personnel, the inspector was informed that this condition results when the filter paper runs out; that this frequently occurs on a back shift or over weekends; and that this results in the monitors being out of service until the next day or Monday morning because the paper is changed by instrument technicians. Mr. Potts stated that during periods when the reactor is shutdown and maintenance is being performed in the containment vessel, these monitors are always kept operational. He also stated that he believed RIC-2 provided adequate backup during operation because of the high gaseous activity in the primary coolant.

Saxton's Technical Specifications do not contain a minimum condition for operation for containment vessel air monitors; however, the Technical Specifications do require that these monitors be operational during maintenance activities performed inside the containment vessel.

U. S. ATOMIC ENERGY COMMISSION
REGION I
DIVISION OF COMPLIANCE

Report of Inspection

CO Report No. 146/70-2

Licensee: SAXTON NUCLEAR EXPERIMENTAL CORPORATION
License No. DPR -4
Category C

Dates of Inspection: September 1 - 3 and 23, 1970

Dates of Previous Inspection: April 20 - 23, 1970

Inspected by: R. T. Carlson for 10/2/70
R. J. McDermott, Reactor Inspector (Responsible) Date

R. L. Spessard 10/2/70
R. L. Spessard, Reactor Inspector (Wrote Report) Date

Reviewed by : R. T. Carlson 10/2/70
R. T. Carlson, Senior Reactor Inspector Date

Proprietary Information: None

## SCOPE

A routine, announced visit was made to the Saxton Nuclear Experimental Corporation (SNEC), Saxton, Pennsylvania, on September 1 - 3, 1970 to inspect the 28 Mwt pressurized water reactor and for official turnover to the newly assigned principal reactor inspector, R. L. Spessard, by R. J. McDermott. A subsequent corporate level interview was held at the Saxton facility on September 23, 1970 and was directed toward Provisional Instruction 1800/2, Chapter 1830, "Corporate Level Interviews" and significant observations made during the September 1 - 3, 1970 inspection.

## SUMMARY

Safety Items - No safety items were identified during this inspection.

Noncompliance Items - The following three items of noncompliance were noted, and a form AEC-592 has been issued.

1. Paragraph N.1.a.(2)(c) of the Technical Specifications requires, in part, that minimum qualifications consisting of five years' experience in engineering, operations, and maintenance at nuclear or fossil-fuel power plants or similar facilities, with two years in a responsible supervisory position of such facilities, and qualification as a licensed senior reactor operator, shall be maintained for personnel occupying the position of Supervisor - Reactor Plant Services.

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Contrary to this requirement, the minimum qualifications with regard to experience and licensing were not maintained when Mr. W. E. Potts assumed the responsibilities of Supervisor - Reactor Plant Services on August 28, 1970 following the departure of Mr. J. G. Herbein.

2. Paragraphs N.1.a.(1) and (2) of the Technical Specifications set forth the organization for the project and for the conduct of plant operations in Figures N.1.a.(1) and (2) which include, in part, test engineer position(s).*

   Contrary to these requirements the test engineer position(s) are vacant.

3. Paragraph I.8.d. of the Technical Specifications requires, in part, that the radioactive concentration of gaseous releases as measured by the radiation monitor in the ventilation duct ahead of the stack fan shall not exceed an instantaneous concentration of $1 \times 10^{-3}$ uCi/cc, excluding I-131, when averaged over a 15 minute period.

   Contrary to this requirement, the stack monitor indicated a radioactive concentration in excess of $1 \times 10^{-3}$ uCi/cc for a period of 3 - 4 hours during an accidental gaseous release occurring on May 14, 1970.

Unusual Occurrences -

1. Section E.1. of this report contains information concerning the separating of the sensing lines to the pressurizer level indicator D/P cell during primary system heatup.

2. Section I.2. of this report contains information concerning the shearing of the 20 ton bridge crane cable.

3. Section O. of this report contains information concerning a dropped irradiated fuel subassembly.

4. Section Q.4. of this report contains information concerning two unplanned releases of gaseous activity to the environs.

Status of Previously Reported Problems - None

Other Significant Items -

1. An IBEW strike involving Saxton's operators and technicians occurred during the period May 27 - June 20, 1970. (Section C.)

2. The licensee has conducted an emergency preparedness drill since the last inspection. (Section D.1.)

3. Saxton's primary coolant activity reached its highest level, 180 uCi/cc, just prior to shutdown for fuel inspection. (Section E.2.)

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*Also, Section 301.B.2 of the Final Safeguards Report specifies two test engineer positions.

4. Fuel pin inspection was conducted at Saxton by the Westinghouse Mobile Fuel Evaluation Team and fuel pin failures were identified. (Section G)

5. Saxton's containment bridge 20 ton crane has been inspected and load tested since the last CO inspection. (Section I.2.)

6. The licensee reported errors made in calculating Noble gas activity concentrations and has corrected his procedures and gaseous releases for calendar year 1970. (Section Q.3.)

Management Interview - September 3, 1970 - Messrs. Swift and Goodman represented SNEC during the management interview held at the conclusion of the inspection. CO was represented by Messrs. McDermott and Spessard.

1. The basis for not correcting past liquid radioactive waste releases due to calculation errors was discussed. The inspectors pointed out that there were no records to indicate that liquid radioactive waste was analyzed for dissolved gaseous activity prior to release. Normal procedures call for boil down of samples prior to analysis. Mr. Swift explained that a substantial portion of the dissolved gases is evolved during the normal processing of liquid radioactive waste and this portion is collected in gas decay tanks. However, he stated that an analysis would be made to substantiate the basis for not correcting past liquid radioactive waste releases.

2. The recent significant organization changes at the Saxton facility were discussed. The inspectors pointed out that the qualifications of Mr. Potts, Supervisor - Reactor Plant Services, did not meet the minimum requirements for this position as set forth in the Saxton Technical Specifications, and that the vacancies in the test engineer positions did not meet the project organization requirements as set forth in the Saxton Technical Specifications. Mr. Swift stated that these changes had been discussed with DRL during a May 6, 1970 meeting and that DRL indicated that although the qualifications of Mr. Potts were marginal, he was acceptable. Mr. Swift further indicated that recruitment for the test engineer positions was in progress but had no idea when these positions would be filled. The inspectors informed Mr. Swift that further compliance action would probably be forthcoming.

3. The dropped fuel subassembly occurrence on August 6, 1970 was discussed. The inspectors pointed out that their review of the On-Site Safety Committee minutes (last meeting on August 20, 1970) indicated that this occurrence was not reviewed by the committee and that according to the Saxton Technical Specifications, plant operations to detect potential safety hazards are to be reviewed by the committee. Mr. Swift stated that the committee had not reviewed this occurrence, but would do so at the next scheduled meeting. The inspectors expressed their concern at the apparent lack of attention given to this potentially hazardous occurrence by Saxton management personnel.

4. The unscheduled releases of gaseous activity to the environs which occurred on May 14, 1970 and August 26, 1970 were discussed. The inspectors pointed out that records indicated that during the releases the stack monitor (RIC-3) was pegged full scale for a period of 3 - 4 hours for the May release and a period of about 5 minutes for the August release. The inspectors further stated that the calibration curve for RIC-3 indicated that when RIC-3 is pegged ( $>$ 1000 cps) the Saxton Technical Specification limit for an instantaneous concentration of $1 \times 10^{-3}$ uCi/cc, excluding I-131, when averaged over a 15 minute period is being exceeded. The inspectors pointed out their reservations concerning the ability of the stack monitor to detect accurately gaseous activity concentrations in the range of $10^{-3}$ uCi/cc and higher due to its limited range and therefore had reservations as to exactly what concentration was actually released. Mr. Swift stated that the calibration curve for RIC-3 presently in use was the original calibration curve made in 1962 and that subsequent calibration checks indicated that the original calibration was conservative and therefore was still being used. Mr. Swift also pointed out that during the May release gas samples taken at the stack indicated that the activity was predominantly Xe-133 and the concentration was $1.2 \times 10^{-5}$ uCi/cc.* Mr. Swift stated that the calibration of the stack monitor would be varified, but if the calibration was found to be conservative, it would probably be left as is. It was the inspectors' position that an accurate calibration was necessary and that utilizing a monitor with a higher response capability should be considered. Mr. Swift was informed that further compliance action would probably be forthcoming.

5. The condition of the carbon steel piping and valves in the Radioactive Waste Disposal Facility (RWDF) was discussed. The inspectors expressed their concern about the present integrity of both the gaseous and liquid waste disposal systems in light of the two recent unscheduled releases of gaseous activity. Mr. Swift stated that an active maintenance program is in progress on the gaseous waste disposal system which consisted of inspection, repair, and testing. He stated that this system was about 50% complete and that valve parts had been ordered. The inspectors indicated that the progress of the maintenance efforts would be followed in future inspections and that they would look for a program of scheduled preventive maintenance on both the gaseous and liquid waste disposal systems. Mr. Swift stated a program would be developed and implemented by the next inspection.

6. The recently identified fuel pin failure was discussed. The inspectors asked Mr. Swift to consider the reportability of this information to DRL, since as a licensing body DRL was interested in fuel cladding performance of the various types of fuels they had approved for use in the reactor. Mr. Swift stated that this matter would be considered.

7. The significance of C-14 activity based on studies by the Public Health Service was discussed. The inspectors asked if the licensee would consider attempting to identify the presence of C-14 activity. Mr. Swift stated that an analysis to determine the presence of C-14 activity would be considered.

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*Inspectors' Note - Sample taken late in release period.

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Management Interview - September 23, 1970 - Messrs. Montgomery and Swift represented SNEC during the interview. Messrs. Goodman, Potts, Beale, Reid, and Pekar from SNEC were also in attendance. CO was represented by Messrs. Carlson, McDermott and Spessard.

Mr. Carlson stated that the purpose of the meeting was two-fold: (1) to meet the CO program objective of holding routine, periodic get-togethers on the corporate level to review matters of mutual interest including, as appropriate, an updating in the areas of regulatory philosophy, practice and organization, and any necessary clarification of the CO role and inspection program; and (2) to discuss the results of the last inspection.

1. Mr. Carlson reviewed the current organization and functions of the Regulatory, with emphasis on DRL and CO, with respect to the various roles performed during the life of a typical facility starting with the initial submittal of an application for a construction permit through normal commercial operation. Special attention was given to the changes made in the CO organization and program. Areas covered included: type, frequency, scope, and depth of CO inspections; inspection techniques; basis for inspections; inspector qualifications; use of consultants; methods and types of enforcement actions; management meetings; emphasis on QA/QC and the applicability of proposed Appendix B of 10 CFR 50 to operating plants; and the requirements and intent of 10 CFR 50.59. It was emphasized that CO inspections are performed on a sampling basis and do not replace the need for the performance of comprehensive audits of operations by licensee management. The requirement that the licensee be able to demonstrate compliance with applicable regulations and the related CO need for access to information were discussed. The philosophy behind typical incident reporting requirements including the concern as to the possible applicability of a particular problem to other facilities, was emphasized.

2. Mr. Carlson stated that the results from the September 1 - 3, 1970 inspection were generally satisfactory although it may not always be apparent since inspectors tend to be problem oriented. He noted that the following observations were made.

   a. Several items of a followup nature had been satisfactorily accomplished by the licensee. (Sections D.1., I.2., and Q.3.)

   b. Several new items had been identified and discussed at the September 3 exit interview. These items (Paragraphs 1, 3, 5, 6 and 7 of Exit Interview and Section I.1.) were again discussed individually with the licensee during this meeting to ensure a common understanding of the proposed resolutions. The licensee's proposed resolutions remained as previously stated to the inspectors. Mr. Swift reported that inspection and repair of the piping and valves in the gaseous waste disposal facility have been completed and that inspection and repair of the piping and valves in the liquid waste disposal system have been initiated. (Paragraph 5 of Exit Interview). Mr. Montgomery reported that all information available to him concerning the inspection of the failed fuel pins had been reported to DRL during subsequent telephone conversations and that this information would appear in the semi-annual report to DRL as required by License DPR-4. He further stated that he did not know whether or not Westinghouse would submit a detailed report of the inspection to DRL because of the proprietary information involved. Mr. Carlson stated that DRL understands the nature of proprietary information and handles it quite often. Mr. Montgomery requested the aid of Compliance in obtaining additional information concerning the measurement of C-14. (Subsequently communicated to Mr. Swift by Mr. Spessard via telecon.)

c. The two items of noncompliance pertaining to plant organization were discussed at length both during and subsequent to this meeting.* Factored in were the results of the discussions held on this subject (plant organization) during the May 6, 1970 DRL-Saxton meeting, including the mechanism by which these issues would be resolved by the licensee. In summary, the licensee (Mr. Montgomery) has agreed that formal resolution of these issues is in order, and proposes to accomplish it as follows:

   (1) Test Engineers - To submit a proposed change to the Technical Specifications to allow filling of these positions with already present Westinghouse employees.

   (2) Qualifications of Mr. Potts - To document in a letter to DRL the substance of the May 6 discussions, including their justification for assigning Mr. Potts to the position of Supervisor - Reactor Plant Services, their proposed short and long range plans to upgrade his qualifications, and their plans including timetable to have Mr. Potts become a licensed senior reactor operator.

d. The item of noncompliance pertaining to the May 14, 1970 accidental gaseous release was also discussed. The licensee (Messrs. Montgomery and Swift) stated that the following actions had been taken:

   (1) Improved the integrity of the gaseous waste disposal system by inspection and repair of 100% of the piping and valves in the system.

   (2) Reviewed the venting procedures for the surge tank and considered possible isolation procedures.

   (3) Routine pressure testing of the vacuum regulating valves and the gas compressor system is being performed.

   (4) Reviewed possibilities of expanding the range of the stack monitor; however, this doesn't appear to be feasible.

   It was Compliance's position that action (4) above should be pursued further because the licensee could not adequately measure a burst type gaseous release. Mr. Montgomery stated this would be reviewed further.

Mr. Montgomery was advised that a form AEC-592 would be issued regarding the three items of noncompliance discussed.

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*Post-meeting discussions included telecons between CO (Mr. Carlson) and DRL (Messrs. Schemel and Woodruff) and DRL and Saxton (Mr. Montgomery) on 9/23 and 24/70.

## DETAILS

A. Persons Contacted:

SNEC

C. R. Montgomery, President (Corporate Level Interview Only)
R. W. Swift, Nuclear Plant Superintendent
D. A. Goodman, Supervisor of Operations and Testing
W. E. Potts, Acting Supervisor of Reactor Plant Services
K. E. Beale, Radiation Protection Engineer
G. Reid, Radiochemist
S. Pekar, Maintenance Foreman

B. Administration and Organization

1. Training

Two senior reactor operators have been licensed at the Saxton facility since the last inspection. Mr. Swift informed the inspectors that one reactor operator and one senior reactor operator are currently in training and will take license examinations this fall. Westinghouse presently has 10 customer personnel in training at Saxton.

2. Personnel Changes

Mr. Swift informed the inspectors of the following significant personnel changes:*

a. Mr. E. A. Liden, Supervisor of Reactor Plant Services left on May 22, 1970, and was replaced by Mr. J. G. Herbein, Supervisor of Operations and Testing. Mr. D. A. Goodman, Test Engineer, vacated his position to assume the responsibilities of supervisor of operations and testing on the same date.

b. Mr. J. G. Herbein, Supervisor of Reactor Plant Services, left on August 28, 1970, and was replaced by Mr. W. E. Potts, Test Engineer and a senior reactor operator in training. Mr. Potts spent six years in the U. S. Navy, the last two of which were on the Nuclear Submarine George Washington in the Inertial Navigation Group where he supervised 12 - 15 personnel. This was a non-nuclear duty position. He worked for three months as a student engineer at Saxton during the summer of 1969 and after receiving a BS degree in electrical engineering from Pennsylvania State University in March of 1970, he returned to Saxton and assumed the responsibilities of a test engineer. The qualifications of Mr. Potts are not in accordance with the minimum requirements for the position of supervisor of reactor plant services as set forth in Section N.1.a.(2)(c) of the Saxton Technical Specifications which requires in part five years' experience in engineering, operations and maintenance at nuclear or fossil-fuel power plants or similar facilities, with two years in a responsible supervisory position of such facilities and qualification as a licensed senior reactor operator.

*Previously identified in CO Report No. 146/70-1.

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c. With the recent promotions of Messrs. Goodman and Potts, the test engineer positions as specified in Section 301, paragraph B.2. of the Saxton Final Safeguards Report and Sections N.1.a.(1) and (2) and Figures N.1.a.(1) and (2) of the Saxton Technical Specifications are presently vacant. Mr. Swift stated recruiting efforts to fill these vacancies are presently in progress.

d. A new position of maintenance foreman, who reports to the supervisor of reactor plant services and is equivalent in level to the radiation protection engineer and the radiochemist was established at Saxton on August 17, 1970 to relieve the supervisor reactor plant services of direct maintenance supervision duties. Mr. S. Pekar, Senior Instrument Technician, is filling this position. Mr. Pekar has 15 years of maintenance experience with instruments which includes both hydraulic and pneumatic types. He came to Saxton in 1962 as an instrument technician and since 1966 has been senior instrument technician, a position which has supervisory duties.

e. Mr. G. Reid, the site radiochemist, was granted a one-year draft deferment and Mr. Swift does not anticipate Mr. Reid's being drafted in the future due to his age. A new radiochemist, Mr. R. Walton joined Saxton in June of 1970 and was recruited as a replacement for Mr. Reid. Mr. Walton recently graduated from Juniata College with a BS degree in chemistry.

f. Since the last inspection five licensed reactor operators have departed the Saxton facility. Four operators went to Three Mile Island and the other operator went to the General Electric Company. There are presently five licensed senior reactor operators and six licensed reactor operators on shift duty. The inspectors concluded that the licensee was in accordance with his Technical Specifications which require two licensed reactor operators at the facility at all times.

The inspectors discussed the noncompliance aspects of the personnel changes enumerated in paragraphs b. and c. above at the exit interview. Mr. Swift stated that their personnel changes were previously discussed with DRL in a May 1970 meeting and that DRL accepted the qualifications of Mr. Potts. The inspectors were informed that the licensee would submit a letter to DRL in accordance with paragraph 3.E.(2) of license No. DPR-4 reporting the recent personnel changes.*

C. Operations

The inspectors reviewed the information in the operation log book, the On-Site Safety Committee minutes, and the SNEC Safety Committee minutes for April 20 - September 1, 1970. In this review the inspectors noted that there was one planned shutdown and no unintentional scrams for this period. Plant operations are summarized as follows:

*Subsequently sent to DRL on September 8, 1970.

1. May 14, 1970

Routine operation which had been maintained since the last inspection was curtailed when the plant was shut down for the scheduled fuel pin inspection by the Westinghouse Mobile Fuel Evaluation Team. At approximately 40 minutes after shutdown an unplanned release of radioactive gases occurred while venting gases from the primary coolant system to the gas compressor system. The licensee reported that an estimated 7.32 curies of gaseous activity (predominantly Xe-133) were released to the environs over a four-hour period.* (See Sections P.3 and Q.4 of this report)

2. May 27 - June 20, 1970

An IBEW strike against the Pennsylvania Electric Company occurred which involved the operators and technicians at the Saxton facility. Mr. Swift stated that there were no significant effects on plant operation or maintenance as a result of the strike due to the fact that cold, shutdown, depressurized conditions existed in the plant, and the five removable subassemblies which were scheduled for inspection had been removed from the core and placed in the spent fuel storage rack.

3. June 26 - August 5, 1970

Fuel pin inspection was conducted on the five removable subassemblies by the Westinghouse Mobile Fuel Evaluation Team. One or both of sister fuel pins from subassembly 504-4-33 were confirmed as leakers. (See Section G of this report.)

4. August 6, 1970

Fuel subassembly 504-4-25 was dropped a distance of approximately 12 inches while being loaded into the spent fuel storage rack following its inspection. Buckling damage to the subassembly can occurred which precluded its reloading into the core. (See Section O of this report.)

5. August 19, 1970

Following completion of core loading and conoseal installation, primary system heatup was initiated utilizing pump heat and pressurizer heaters.

6. August 20, 1970

Separation of both 3/8 inch sensing lines to the pressurizer level indicator D/P cell occurred during primary system heatup at 2000 psig and 636° F in the pressurizer. Excess flow check valves functioned as designed and there was no primary leakage. (See Section E.1. of this report.)

7. August 24, 1970

After completion of the hot functional primary system pressure test at 2285 psig and 512° F (50 psig above operating pressure), the reactor was made critical and flux mapping began.

*Letter to DRL dated May 18, 1970.

8. August 26, 1970

An unplanned release of radioactive gases occurred while operating personnel were switching gas compressors to locate and isolate a minute leak in the RWDF gas compressor system. The licensee reported that an estimated 0.034 curies of gaseous activity (predominantly Xe-133 and Xe-135) were released to the environs over a five-minute period.* (See Sections P.4. and Q.4. of this report.

9. August 31, 1970

Load cycling operations were initiated and were in progress during the inspection. This operation consisted of cycling between 100% and 40% of full power with 6 cycles being performed on the day and swing shifts during a 5-day week. Mr. Swift stated that load cycling operations to test fuel performance and Westinghouse customer training would continue until the mid-life fuel inspection which is scheduled for November 1970.

D. Facility Procedures

1. The inspectors verified, in discussions with Mr. Swift that the licensee conducted an emergency preparedness plan drill on August 26, 1970. Mr. Swift stated that the results of the drill were satisfactory and that the drill included phone checks to support agencies. This is a followup item from the last inspection.**

2. The inspectors reviewed the calibration procedure for the Steam Generator Blowdown Monitor (RIC-5) and noted that the last calibration using known source strengths (radioactive primary coolant water) was performed on October 14, 1969. The inspectors also reviewed the calibration curve (uCi/cc vs CPS) for RIC-5. The monitor is given a source check monthly utilizing a Co-60 source and a standard geometry setup. The calibration point is 120 cps and the alarm point is set at 3 cps.

E. Primary Systems

1. Separation of Sensing Lines to Pressurizer Level Indicator D/P Cell

In discussions with Messrs. Swift and Pekar it was learned that the pressurizer channel (LIC-2), which provides signals for continuous level indication, "On" - "Off" high and low level alarms, low-level heater shut-off, and low-level scram, indicated an increase in level from 50% to 100% during primary system heatup (2000 psig and 636° F in pressurizer). The plant was cooled down and investigation revealed that the Tylock fittings in both 3/8 inch stainless steel tubing lines connecting the pressurizer level column to a D/P cell had separated which caused the level indication on LIC-2 to increase to 100%. There was no primary system leakage as a result of this occurrence because the excess flow check valve (Chemiquip Type 50 FM 100) located upstream of the Tylock fitting in both lines, functioned as designed. The separation occurred above the stainless steel

*Letter to DRL dated August 31, 1970.
**CO Report No. 146/70-1.

valve block assembly, which is mounted in the D/P cell cabinet. The arrangement is shown in Figure 1 attached. Prior to system heatup the D/P cell had been removed for calibration by uncoupling the D/P cell from the valve block assembly at the Swaglock fittings. Corrective action taken by the licensee consisted of installing a new valve block assembly, replacing the Tylock fittings with Swaglock fittings, recalibrating the D/P cell, and performing a hot functional primary system pressure (leak) test at 2285 psig and 512° F.

2. Primary Coolant Activity

Examination of records for analyses of primary coolant disclosed that the highest coolant activity, as shown below, occurred on May 14, 1970, just prior to the scheduled shutdown for fuel inspection.

| Total | Degassed | Gaseous |
|---|---|---|
| 180.03 uCi/cc | 16.80 uCi/cc | 163.23 uCi/cc |

Since returning to operation on August 24, 1970, the total coolant activity has decreased to about 14 uCi/cc. The reduction in coolant activity resulted from the removal of failed fuel pins from the core.

F. Reactivity Control and Core Physics

1. Core Power Distribution

The in-core monitoring program was discussed with Mr. Goodman, and based on weekly and monthly Westinghouse reports, Saxton projects ahead their operating power limits (currently 22.80 Mwt) based on peak pellet burnup (MWD/MTM) and peak pellet power (Kw/ft). The inspectors reviewed operating data taken on August 31, 1970 at an equilibrium power of 22.77 Mwt. The data which is summarized below indicates the licensee is operating within the established Technical Specification limits.*

| Assembly | Core Position | Peak Pellet Power | Peak Pellet Burnup |
|---|---|---|---|
| Pu. Loose Lattice | A9 in D-4 | 21.57 Kw/ft | 27,233 ~~25,283.76~~ |
| U Load Follow | A7 in E-3 | 18.34 Kw/ft | 9,828 ~~4,983.21~~ |

2. Control and Safety System

The maximum time from scram initiation to scram completion was reviewed by the inspectors. Maximum time allowed by the Saxton Technical Specifications is 1.5 seconds. This measurement is the total of the control rod drive scram speed and scram circuit response time. The last test of control rod drive scram speed was conducted March 23, 1970. The slowest time recorded was 1.050 seconds for control rod No. 4. Scram circuit response time was last tested July 27, 1970. The slowest response time

*Section G.3.

recorded was 0.225 seconds for the high main coolant temperature scram. The combined time, based on the slowest times, is 1.275 seconds for scram initiation to scram completion, which is within the 1.5 seconds allowed by the Saxton Technical Specifications.*

G. Core and Internals

During the last shutdown the Westinghouse Mobile Fuel Evaluation Team inspected the fuel pins in the five removable subassemblies. The fuel pins were cleaned, the pin diameters measured, and pictures taken with a TV camera. Mr. Swift stated that the general condition of the fuel was good with the exception of sister fuel pins No. 504 and No. 505 in fuel subassembly 504-4-33, a peripheral assembly in core position N-2.** These fuel pins were monitored together and one or both pins were confirmed as leakers. Mr. Swift stated that one pin had blisters and the other had a hole approximately 1/4 inch in diameter on the cladding surface which tapered to approximately 1/8 inch in diameter at some depth into the pin. The exact location of the hole with respect to the experimental fluoride contaminated longitudinal notches was not made available to the inspectors. These fuel pins were subsequently shipped to the Westinghouse Post Irradiation Facility at Waltz Mill, Pennsylvania for an in-depth inspection. The reportability to the AEC (DRL) of the results of these fuel pin inspections was discussed at the exit interview.

I. Auxiliary Systems

1. Piping Modification to Accumulator on Charging Pump Discharge Header

The 1 inch, schedule 160, 304 stainless steel seamless pipe which connects the accumulator to the charging pump discharge header has been extended approximately 12 inches by utilizing a 1 inch, schedule 160, 304 stainless steel pipe coupling rated at 4000 psi and a 12 inch length of 1 inch, schedule 160, 304 stainless steel pipe. This modification was made to permit easier access for maintenance. A review of the modification package revealed that the pipe and welding wire used met Code ANSI B31.1 requirements, but the material certificate for the coupling was not available. Mr. Swift stated that the certificate had been requested from Westinghouse but had not been received. The inspectors will pursue this matter during the next inspection. The welding procedure, welder, and welding inspector, which were supplied by Westinghouse, were qualified to applicable code requirements. The welds were L.P. tested in accordance with applicable code requirements and the system was hydrostatically tested to 3750 psig in accordance with the Saxton Final Safeguards Report.*** The design system pressure is 2500 psig.

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*Section N.4.f.(4).

**Fuel pin description given in: Safeguards Report for Saxton Core III, Table 2.1-3; Appendix C - Change Report 18; and in Saxton Technical Specifications, Paragraph F.3.h.(2).

***Section 206.E.

2. Containment Bridge 20 Ton Crane

Mr. Swift informed the inspectors that on July 1, 1970, a new cable (9/16 inch diameter Roebling Wire Rope) was installed on the bridge crane, the limit switches set, and the gear box, gear box to drive motor coupling, drum to gear box coupling, and drum coupling inspected by the factory technical representative. Due to an operator error, the old crane cable was sheared when it was inadvertently allowed to slip off the drum and wrap around the drum axle. The On-Site Safety Committee reviewed this occurrence and recommended that during use the crane be attended by at least two SNEC personnel, one operator and one director.

Mr. Swift stated that Westinghouse's chief crane inspector load tested the 20 ton bridge crane to 111% of design load on July 6, 1970. The inspectors reviewed both the inspection and load test reports and noted that the reports appeared to be complete and in order. The inspectors also noted that the crane inspector recommended replacement of the four lifting slings. Mr. Swift stated that the slings had been ordered. Testing of the 20 ton bridge is a followup item from the last two inspections.*

O. Fuel Handling

Licensee correspondence with DRL** and subsequent discussions with Messrs. Swift, Goodman, and Potts revealed that after inspection by the Westinghouse Mobile Fuel Evaluation Team, fuel subassembly 504-4-25 was dropped approximately 12 inches while being loaded into the spent fuel storage rack. Examination by the licensee revealed that the fuel pins were not damaged, but the 0.019 inch thick perforated stainless steel can showed buckling damage about 4 - 5 inches from the top and 3 - 4 inches from the bottom of the can. It was decided by the licensee that this subassembly would not be reloaded into the core. (See previously referenced letter). The subassembly fell when the support tube separated from the protective sleeve during normal handling. It was determined by the inspectors that this occurrence could have happened at any time during the handling period due to an apparent failure in the locking device and that this irradiated subassembly could have been dropped a distance of up to 14 feet.

An illustration of the mechanical fitup for lifting a fuel subassembly is given in Figure 2. The locking action occurs when the locking screws are turned 90 degrees by an operator after the protective sleeve is lowered into position over the guide tube. The fitup occurs prior to raising the water level above the top of the reactor pressure vessel head in preparation for fuel handling. Mr. Swift stated that a large force is required to turn the locking screws but in this case the operator noted that less force than normal was required during the fitup. Mr. Swift indicated that both screws apparently rotated during subsequent fuel handling, and that the fuel handling was considerably more than normal due to the inspection of the fuel pins. A review of the draft On-Site Safety Committee minutes for the August 20, 1970 meeting indicated that this occurrence was not reviewed by the committee. This matter was discussed at the exit interview.

*CO Report Nos. 146/69-4 and 146/70-1.

**Letter to DRL, dated August 18, 1970, Technical Specification Change Request #37.

P. Radiation Protection

1. Monitoring Equipment

The inspectors reviewed Saxton's surveillance testing records and verified that the radiation monitor circuits had been tested on monthly intervals for the period January - August 1970 and that the semi-annual 24-hour level test to verify the inner tank integrity of the liquid waste storage tanks had been performed and satisfactory results obtained as required by the Saxton Technical Specifications.*

2. Personnel Monitoring

Examination of personnel monitoring records for the period of April - August 1970 disclosed that the highest individual exposure was 1350 mrem in one month and 1785 mrem in one quarter. The inspectors verified that the licensee had determined individual accumulated occupational doses to the whole body as required by 10 CFR Part 20.101 and 20.102.

3. Personnel Monitoring for May 14, 1970 Unplanned Release of Radioactivity**

The licensee reported that the maximum concentration of gaseous radioactivity to which personnel were exposed while in the gas compressor room was $3.3 \times 10^{-4}$ uCi/cc (predominantly Xe-133) and that based on consultation with their medical-radiation consultant and as a precautionary measure, bioassays on all personnel involved were obtained.*** The events of this release were discussed with Messrs. Swift and Beale. It was learned that the highest external exposure received during the release was 10 mrem as indicated by one individual's pocket dosimeter. Bioassays (urine and fecal) for the personnel involved were reviewed by Saxton's medical-radiation consultant, the On-Site Safety Committee, and the SNEC safety committee, and it was determined that there was no evidence of body burden. A review by CO:I of the bioassays did not indicate any significant body uptakes.

4. Personnel Monitoring for August 26, 1970 Unplanned Release of Radioactivity****

The licensee reported a small release of gaseous radioactivity (0.034 Ci of predominantly Xe-133 and Xe-135 over a five minute period) and that there were no personnel exposures.***** In discussions concerning the events of this release with Messrs. Swift and Beale it was learned that radiation readings, smear surveys, and gas samples taken in the gas compressor room were normal. Nasal swipes and smears were taken on personnel in the area and the results were negative. As a precautionary measure, the licensee had bioassays (urine) performed on personnel in the area. The results of the bioassays had not been received from Tracerlab during this inspection and will be reviewed by the inspector during the next inspection.

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*Paragraph N.8.b.
**Inquiry Memorandum No. 146/70-B.
***Letter to DRL dated May 18, 1970.
****Inquiry Memorandum No. 146/70-C.
*****Letter to DRL dated August 31, 1970.

CO Report 146/70-2

Q. Radioactive Waste Systems

1. Liquid Waste

Liquid release records for the period January - August 1970 were reviewed by the inspectors and no indication was found that releases have exceeded applicable limits. Following is a summary of liquid waste releases for this period.

| Beta-Gamma | Tritium |
|---|---|
| 0.007521 Curies | 6.7224 Curies |

2. Gaseous Waste

Gaseous release records for the period January - August 1970 were reviewed by the inspectors and a summary of gaseous waste releases for this period and the time averaged percent of limit are given below:

| Xe-133 & Xe-135 | % of Limit | I-131 | % of Limit |
|---|---|---|---|
| 1708.2 Curies | 68% | 0.011071 Curies | 0.17% |

Release limits of 10 curies/year of I-131 and 3750 curies/year of krypton and xenon are stipulated in the Saxton Technical Specifications. Releases for the period of June - August 1970 totaled about 6 curies and with the removal of the failed fuel pins it would appear that the applicable release limits for calendar year 1970 will not be exceeded.

3. Calculation of Gaseous Activity Concentration

The licensee reported by telephone to CO:I on June 1, 1970 that Westinghouse had discovered errors in the gamma abundance factors being used by Saxton personnel to calculate gaseous activity concentration. In view of this finding Westinghouse reviewed all of Saxton's counting procedures to verify their adequacy for determining activity releases. The licensee stated that gaseous releases for the period January - May 1970 would be corrected and published in the Saxton Monthly Report for May 1970.

In discussions with Messrs. Swift and Reid the inspectors noted that the following corrections had been made to the gamma abundance factors used in the Noble gas counting procedure.

| Isotope | $\gamma$ - Abundance Factor | |
|---|---|---|
| | Old | New |
| Xe-133 | 1.0 | .37 |
| Xe-135 | 1.0 | .91 |
| Kr-87 | 1.0 | .85 |
| Kr-88 | 1.0 | .35 |

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After reviewing the gaseous activity release records and the Saxton Monthly Report for May 1970, the inspectors concluded that the licensee had corrected gaseous releases for the period of January - May 1970, and that after May 31, 1970, all gaseous releases were calculated using the new gamma abundance factors.

The inspectors also noted that past liquid releases had not been corrected by the licensee and that no records were available to substantiate the basis for not correcting these releases. The inspectors questioned the lack of definitive basis on the part of the licensee for not correcting the liquid releases at the exit interview. Mr. Swift stated that an analysis would be performed to substantiate the basis for not correcting past liquid waste releases.

4. Unplanned Gaseous Activity Releases to the Environs

May 14, 1970 Release*

The inspectors reviewed the events of this release with Messrs. Swift, Beale and Pekar. It was noted that during the release the stack monitor (RIC-3) was pegged full scale (reading > 1000 cps) for a period of 3 - 4 hours. Examination of the calibration curve for RIC-3 indicated that a reading of greater than 1000 cps corresponds to a gaseous activity concentration of greater than $1 \times 10^{-3}$ uCi/cc. Based on these findings it appeared to the inspectors that the Saxton Technical Specification release limit for an instantaneous concentration of $1 \times 10^{-3}$ uCi/cc, excluding I-131, when averaged over a 15-minute period** was exceeded during this release. This matter was discussed at the exit interview.

The licensee informed the inspectors that further investigation of No. 1 gas compressor vacuum regulator valve (Fisher Governor Type Y600) disclosed that, contrary to the information given in their report to DRL, the diaphragm had not ruptured but rather the adjusting rod for setting desired system vacuum was found adjusted in such a manner that the center hole in the diaphragm had lost its seal and thus provided the leakage path.

The inspectors also verified that the release of an estimated 7.32 curies of gaseous activity which was predominantly Xe-133, as reported by the licensee, was calculated using the old gamma abundance factor for Xe-133 of 1.0, and based on the new gamma abundance factor for Xe-133 of 0.37 would yield a release of an estimated 19.76 curies.

The release was reviewed by the On-Site Safety Committee and the SNEC Safety Committee and the following precautionary actions have or will be taken:

*Inquiry Memorandum No. 146/70-B.
**Section I.8.d.

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a. The surge tank venting procedure was reviewed, however no changes were made because the release was caused by mechanical failure.

b. Routine pressure testing of vacuum regulating valves and the gas compressor system.

c. Review proposed isolation procedures.

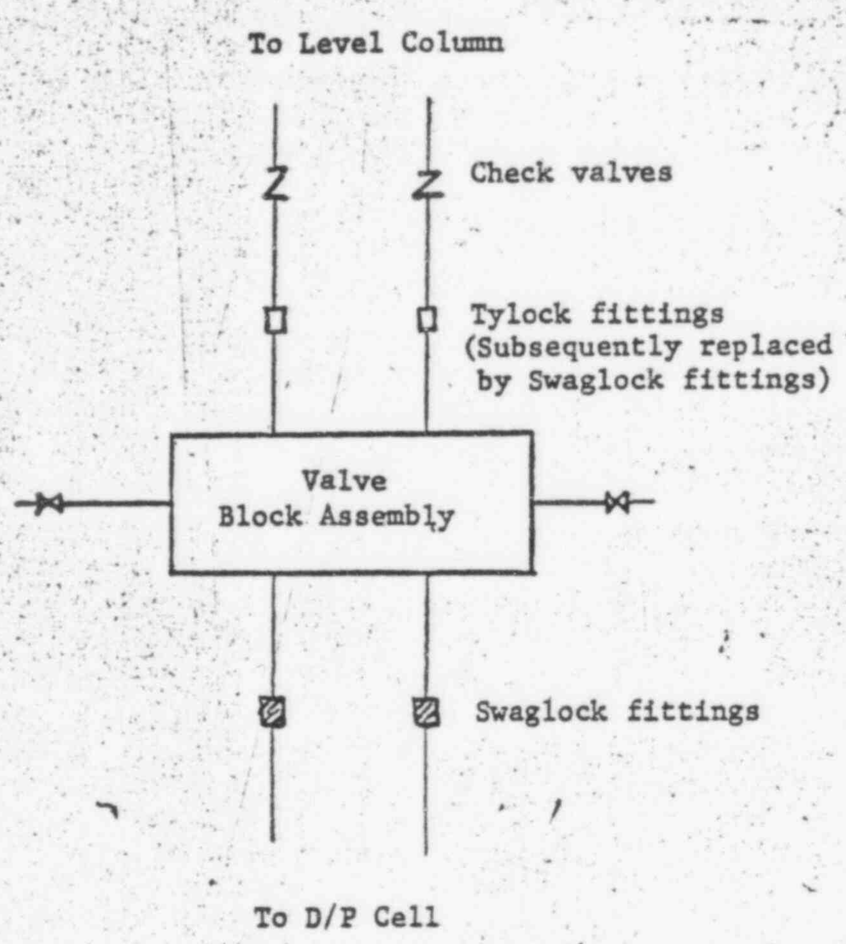
d. Instruction of SNEC personnel in the proper use of a Scott Air Pack.

August 26, 1970 Release*

The events of this release were discussed with Messrs. Swift, Goodman, Pekar and Beale. It was noted that for approximately five minutes during this release RIC-3 recorded a reading in excess of 1000 cps or a concentration greater than $1 \times 10^{-3}$ uCi/cc. Based on the total activity released, time duration of the release, and dilution factor in the stack, the inspectors concluded that the Technical Specification instantaneous release limit of $1 \times 10^{-3}$ uCi/cc, excluding I-131, when averaged over a 15-minute period probably was not exceeded.

The inspectors concluded that the release occurred as reported by the licensee, but noted that leakage back through gas compressor systems involved passage through six check valves prior to release. The RWDF (both gaseous and liquid systems), which is composed of carbon steel valves and piping, is subject to corrosive attack from liquid waste, especially boric acid. The inspectors reviewed maintenance records which indicated that six check valves and six globe valves in the gas compressor system had been inspected and repaired where possible with deficiencies noted. The licensee indicated that replacement parts for defective valves had been ordered. The general condition of the RWDF was discussed at the exit interview.

*Inquiry Memorandum No. 146/70-C.




POOR ORIGINAL

D/P CELL CABINET ARRANGEMENT

Figure 1

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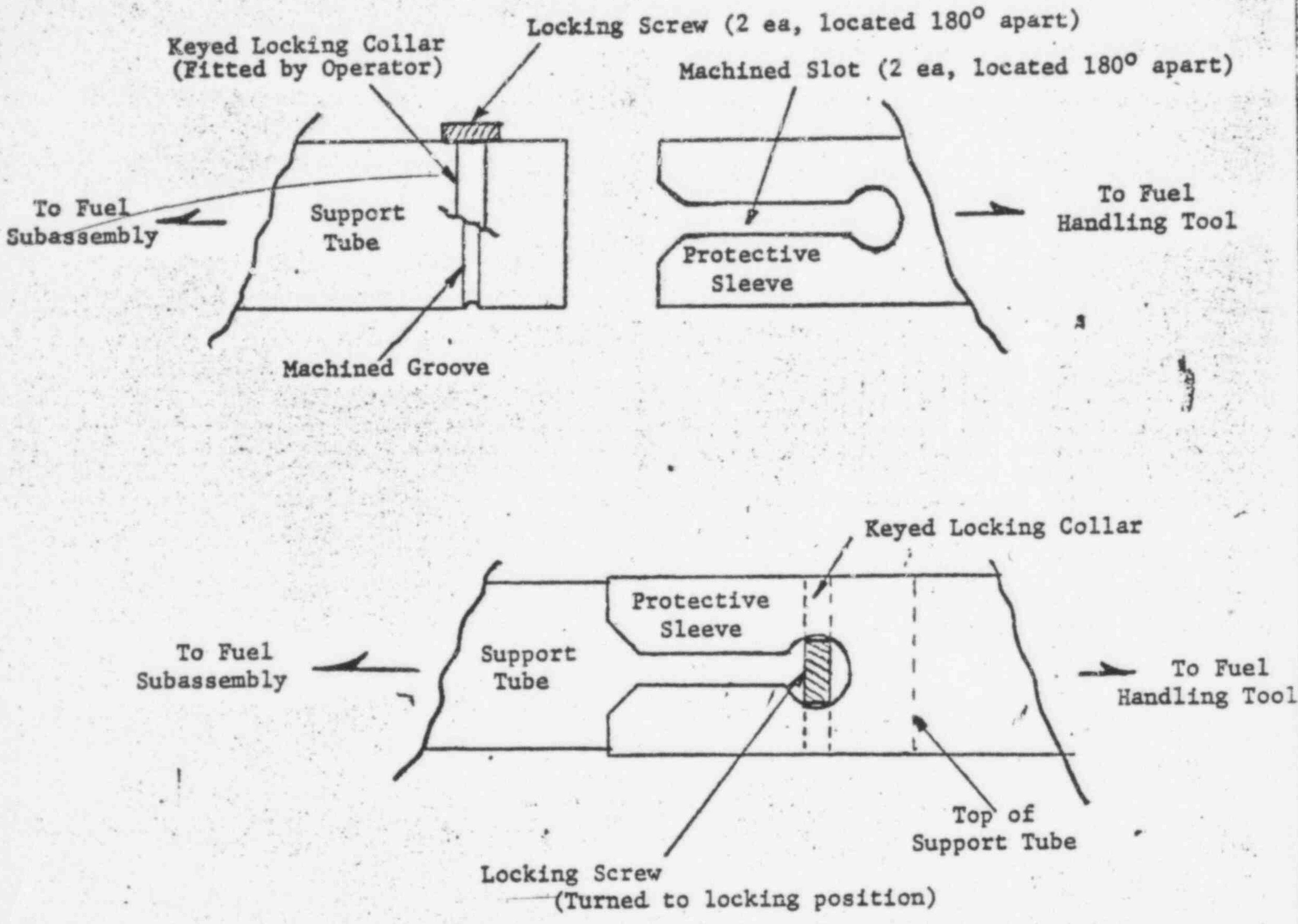


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FITUP FOR LIFTING FUEL SUBASSEMBLY

Figure 2