NUCLEAR REACTOR LABORATORY
THE UNIVERSITY OF MICHIGAN

Michigan Memorial-Phoenix Project
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March 30, 1994

U.S. Nuclear Regulatory Commission
Region III
Regional Administrator
801 Warrenville Road
Lisle, Illinois 60532-4351

Re: License R-28
Docket 50-2

Dear Sir:

The enclosed REPORT ON REACTOR OPERATIONS for the period January 1, 1993 to December 31, 1993 is submitted to comply with Section 6.6 of the Ford Nuclear Reactor Technical Specifications.

Sincerely,

Ronald F. Fleming

Ronald F. Fleming
Director
Michigan Memorial-Phoenix Project

xc: United States Nuclear Regulatory Commission
Document Control Desk
Attn: Theodore S. Michaels, Project Manager
Standardization and Non-Power Reactor Project Directorate
Division of Reactor Projects III, IV, V, and Special Projects
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Washington, D.C. 20555

American Nuclear Insurers

FNR Safety Review Committee
FNR Health Physicist
FNR Control Room

IE47
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Enclosure (1)

050076

Subscribed and sworn before me this 28th day of March, 1994, a Notary Public in and for Washtenaw County, Michigan.

Kathe C. Blair

NOTARY PUBLIC
My Commission expires Feb. 17, 1992.

REPORT OF REACTOR OPERATIONS

January 1, 1993 to December 31, 1993

FORD NUCLEAR REACTOR

MICHIGAN MEMORIAL - PHOENIX PROJECT

THE UNIVERSITY OF MICHIGAN

ANN ARBOR

March 1994

Prepared For

The U.S. Nuclear Regulatory Commission

ABSTRACT

Technical Specifications for the Ford Nuclear Reactor (FNR) require the annual submission of this review of reactor operations to the U.S. Nuclear Regulatory Commission (NRC).

The 1993 reactor schedule of ten days of continuous operation at licensed power of two megawatts followed by four days of shutdown resulted in 5,682.1 reactor operating hours, 4,679.8 operating hours at full power, 9,959.5 accumulated megawatt hours, and an overall reactor availability of 65 percent for the calendar year.

Eight regular fuel elements and no control rod fuel elements were retired from operation this year.

There were two reportable occurrences in 1993: Number 17, operation of the Ford Nuclear Reactor at 2.3 Mw for between 10 and 11 minutes; and Number 18, release of low level radioactive water from the Ford Nuclear Reactor building to drain tiles around the foundation of the building.

There were 20 unscheduled reactor shutdowns during 1993.

There were no radioactive effluent releases above 10CFR20 limits. The maximum whole body exposure received by an individual at the facility was 1.27 rem. The cumulative "deep" whole body exposure for the workers at the facility was 12.79 rem.

FORD NUCLEAR REACTOR

Docket No. 50-2
License No. R-28

REPORT OF REACTOR OPERATIONS

January 1, 1993 - December 31, 1993

This report reviews the operation of the University of Michigan's Ford Nuclear Reactor for the period January 1 to December 31, 1993. The report is to meet the requirement of Technical Specifications for the Ford Nuclear Reactor. The format for the sections that follow conforms to Section 6.6.(1) of Technical Specifications.

The Ford Nuclear Reactor is operated by the Michigan Memorial-Phoenix Project of the University of Michigan. The Project, established in 1949 as a memorial to students and alumni of the University who died in World War II, encourages and supports research on the peaceful uses of nuclear energy and its social implications. In addition to the Ford Nuclear Reactor (FNR), the Project operates the Phoenix Memorial Laboratory (PML). These laboratories, together with a faculty research grant program, are the means by which the Project carries out its purpose.

During 1993, as in previous years, the operation of the Ford Nuclear Reactor has provided major assistance to a wide variety of research and educational programs. The reactor provides neutron irradiation services and neutron beamport experimental facilities for use by faculty, students, and researchers from the University of Michigan, other universities, and industrial research organizations. Reactor staff members teach classes related to nuclear reactors and the Ford Nuclear Reactor in particular and assist in reactor-related laboratories.

Tours are provided for school children, university students, and the public at large as part of a public education program. During 1993, 1119 people participated in 96 tours.

The operating schedule of the reactor enables a sustained high level of participation by research groups. Continued support by the Department of Energy through the University Research Reactor Assistance Program [Contract No. J-KT-0300-000 (DE-AC02-76ER00385)] and the Reactor Facility Cost Sharing Program (Contract No. DE-FG07-80ER10724) has been essential to maintaining operation of the reactor facility.

1. OPERATIONS SUMMARY

In January, 1966, a continuous operating cycle was adopted for the Ford Nuclear Reactor at its licensed power level of two megawatts. The cycle consisted of approximately 25 days at full power followed by three days of shutdown maintenance. In June, 1975, a reduced operating cycle consisting of ten days at full power followed by four days of shutdown maintenance was adopted. A typical week consisted of 120 full-power operating hours. In July, 1983, the reactor operating schedule was changed to Monday through Friday at licensed power and weekend shutdowns. Periodic maintenance weeks were scheduled during the year. In January, 1985, a cycle consisting of four days or 96 full-power operating hours per week at licensed power followed by three days of shutdown maintenance was established in order to eliminate the periodic shutdown maintenance weeks needed in the previous cycle. Beginning July 1, 1987, the reactor operating cycle returned to ten day operation at full power followed by four days of shutdown maintenance. Calendar year 1993 began with cycle 350 and ended with cycle 362. A cycle covers four weeks; two of the ten day - four day sequences.

The reactor operates at a maximum power level of two megawatts which produces a peak thermal flux of approximately 2×10^{13} $n/cm^2/sec$. An equilibrium core configuration consists of approximately 40, 19.75% enrichment, plate-type fuel elements. Standard elements contain 167 gm of U235 in 18 aluminum clad fuel plates. Control elements, which have control rod guide channels, have nine plates and contain 83 gm of U235. Overall active fuel element dimensions are approximately 3"x3"x24".

Fuel elements are retired after burnup levels of approximately 35-40% are reached. Fuel burnup rate is approximately 2.46 gm U235/day at two megawatts.

1.1 Facility Design Changes

None

1.2 Equipment and Fuel Performance Characteristics

Reactor equipment and fuel exhibited no abnormal characteristics. Replacement of expended fuel elements resulted in an annual use of eight standard fuel elements and no control fuel elements.

Sixteen new fuel elements were received: twelve standard and four control.

There were no spent fuel shipments.

1.3 Safety-Related Procedure Changes

Safety-related procedures are those associated with operation, calibration, and maintenance of the primary coolant, the reactor safety system, the shim-safety rods, all scram functions, the high temperature auto rundown function, and the pool level rundown.

<u>Operating Procedure 101 - Reactor Start-up</u>

1. Clarified the interface between the reactor startup procedure and performance of a reactor calorimeter measurement to determine reactor power at the beginning of each reactor operating cycle.

2. Proper Response Range of neutron channels at 500 kw check tightened.

3. Startup checklist modified to perform proper checks on the upgraded safety system.

<u>Calibration and Maintenance Procedure 201 - Shim-Safety Rod Calibration</u>

1. Procedure modified for full length rod calibrations; previously upper-half-length calibrations were performed with the assumption that the lower half of the rods were symmetric with the upper half.

<u>Calibration and Maintenance Procedure 203 - Rod Release-Drop Time Measurement</u>

1. Procedure modified to accommodate rod release and drop timer associated with upgraded safety system and above surface magnets.

<u>Calibration and Maintenance Procedure 205 - Safety Channel A and B Calibration</u>

1. Procedure completely revised to accommodate upgraded safety system.

<u>Calibration and Maintenance Procedure 206 - Safety System Period Channel C Calibration</u>

1. Procedure completely revised to accommodate upgraded safety system.

<u>Calibration and Maintenance Procedure 218 - Magnet Power Supply Calibration</u>

1. Procedure prepared for shim-safety rod magnet power supply which is separated from the safety channel. In the old system, magnet power supplies were incorporated within the safety channels chassis.

Calibration and Maintenance Procedure 301 - Shim-Safety Rod Inspection

1. Rod removal and inspection procedure was modified to accommodate above surface magnet system.

1.4 Maintenance, Surveillance Tests, and Inspection Results as Required by Technical Specifications.

Maintenance, surveillance tests, and inspections required by Technical Specifications were completed at the prescribed intervals. Procedures, data sheets, and a maintenance schedule/record provide documentation.

1.5 Summary of Changes, Tests, and Experiments for Which NRC Authorization was Required.

None

1.6 Operating Staff Changes

The following reactor operations staff changes occurred:

New Employees	Date of Hire
Bernard Ducamp (Assistant Manager for Operations)	May 24, 1993
Phillip Heuker (Reactor Operator)	July 6, 1993

Resigned or Retired	Date:
Clifford Slay (Senior Reactor Operator)	May 3, 1993
Gary M. Cook (Assistant Manager for Operations)	October 29, 1993

1.7 Reportable Occurrences

1.7.1 Reportable Occurrence No. 17
Operation of the Ford Reactor at 2.3 Mw for Between 10 and 11 Minutes

Description

On Wednesday, March 24, 1993, during a reactor startup, calorimetric power level determination, and escalation to full power at the beginning of a ten-day operating cycle, power was increased to 2.3 Mw.

The shift crew had completed a routine reactor checkout and startup to an indicated reactor power of 100% on the Linear Level system 1 Mw range. The reactor was in automatic control with the control system setpoint at 100%. A calorimetric determination of reactor power was conducted utilizing Operating Procedure 106, Power Level Determination. Actual thermal power was determined to be 1.156 Mw.

At this point, the shift crew should have reduced the automatic control system setpoint and Linear Level indication to 86% (100%*[1.0/1.156]) to reduce thermal power to 1 Mw.

At 0705, the Shift Supervisor directed power to be raised to an indicated power to 2 Mw (100% on the 2 Mw Linear Level range). When indicated power reached 2 Mw, actual thermal power was 2.3 Mw.

At approximately 0715, the Assistant Reactor Manager for Operations arrived. After a review of the calorimeter data, he immediately ordered the reactor to be returned to 1 Mw indicated power. At 1 Mw, corrective actions were taken to adjust indicated power on the neutron measurement channels to actual thermal power.

Corrective Action

Operating Procedure 106 was modified and retitled, Power Level Determination and Increase to Full Power. Dry runs were conducted by the three shift crews. The modifications were approved by the Safety Review Committee on April 6, 1993, and the procedure was utilized during the routine reactor startup on April 7, 1993, under the supervision of reactor management.

The revised procedure specifically requires that the automatic control system setpoint be adjusted to 1 Mw before going to full power, and that neutron detection chambers be adjusted to within their prescribed ranges at 1 Mw, if chamber adjustments are necessary. A review of the calorimetric power level determination and chamber adjustments must be conducted by reactor management before power is increased to 2 Mw.

The senior reactor operator who was the Shift Supervisor on the shift crew has been removed from all licensed duties and is no longer employed by the facility.

1.7.2 <u>Reportable Occurrence No. 18</u>
<u>Release of Low Level Radioactive Water from the Ford Nuclear Reactor Building to Drain Tiles Around the Foundation of the Building</u>

Description

The Ford Nuclear Reactor conducts periodic inventory measurements of reactor pool water because of known leakage through the concrete walls of the pool. The specific procedure used is CP-502, Pool Water Surveillance. The purpose of the surveillance measurements is to ensure that reactor pool water is not leaking into the ground under the reactor building. A decrease in pool level is the measure of loss of pool water. The loss is accounted for by measuring water collection in three sumps called the cold sump, the hot sump, and the thermal column trench. Additional water losses result from pool surface evaporation and from evaporation of leakage water.

During a pool water surveillance conducted on April 27, 1993, the measured pool water losses exceeded the decrease in pool level by approximately 200 gallons. That is, 200 more gallons were accounted for than the receding pool level indicated. All previous surveillances had resulted in only nominal differences between decreasing pool level and collection and evaporation. The surveillance was repeated on June 16, 1993 on the assumption that a measurement error was made in the April 27 surveillance. The June 16 surveillance resulted in approximately the same "excess" of water. An investigation revealed that a check valve in the discharge line from the cold sump was allowing water to leak back into the sump. In the data taking process, this leak back was interpreted as additional water that was collected in the cold sump.

The cold and hot sumps were inspected. The check valve on the discharge of the hot sump did not leak back. It was not possible to immediately replace the cold sump check valve. The surveillance, which is required semiannually, had been done on time, but conclusive results had not been obtained. The cold sump pump and a discharge valve on the pump outlet were secured and properly tagged out on June 23, 1993. A cross connect between the two sumps allowed the cold sump to fill up to a certain level and then spill over into the hot sump. The cold sump was operated in this manner and closely observed for some time to ensure that operation was satisfactory.

Another pool water surveillance was conducted on July 21, 1993, with combined drainage into the cold and hot sumps collected in the hot sump alone. The results of the surveillance showed a net loss of approximately 640 gallons of water over the 46 hours of the measurement. This was interpreted as a possible measurement error.

Over the next few days, the situation was investigated. There were no indications of unexpected water collection, seepage, or leakage anywhere in or around the reactor pool. The pool water make-up was reviewed. Pool water make-up comes from two sources: recycled waste water (the same water that is collected in the cold sump, hot sump, and thermal column trench) and Ann Arbor city water. It was noted that the amount of recycled water available during the month of July was unusually low as a fraction of total reactor pool water make-up when compared to preceding months.

The only change in the water processing systems was putting the cold sump pump out of commission. The cold sump was restored to normal on July 30, 1993. Operation was observed for a few days. For that short time, the balance of recycled make-up to total pool make-up seemed to be restored. Another pool water surveillance was conducted on August 3, 1993. The results were back to normal, that is, within reasonable measurement errors, all water was accounted for. It was clear that water had been lost from the facility and the cold sump seemed the most likely source of the loss.

The cold sump was reviewed in detail. Several drain pipes enter the sump from various floor drains. Drawings were reviewed. One building drawing indicated that a line from the foundation tile drains fed into the cold sump, though the precise details and layout of the drainage system did not exist.

A controlled experiment was performed to validate out leakage. The cold sump was pumped down to the facility retention tanks and isolated. An inflatable seal was inserted in the overflow line to the hot sump. The cold sump was filled with non-radioactive city water to a level above the overflow line, and the level was monitored to determine if water was leaking out of the sump through the tile drain line. The cold sump level did decrease at an equivalent rate of approximately 200 gallons per day until the water reached approximately two inches above the tile drain line. At that time, the level remained steady for an hour of measurements. The leak rate is consistent with the approximately 7,500 gallons of unaccounted for water over the 36 days that the cold sump pump was secured. The fact that the level stopped receding at a point above the line to the drain field indicates that the drain tiles are above the cold sump and the purpose of the drain field line is to permit rain and other water to enter the cold sump rather than erode the building foundation. In addition, the fact that the cold sump level stopped receding indicates that the sump itself does not leak; it is a poured concrete structure.

Consequences of Release

A sample was drawn from the cold sump during reactor operation. Fifteen-minute and three-day analyses were performed with the following results. The short decay detracts from the accuracy of measurement of long-lived nuclides.

	Radioactivity Concentration		
Nuclide	15-Minute	Three-Day	MPC Unrestricted
Na-24	6.35×10^{-3}		2.00×10^{-4}
Mn-56	9.94×10^{-6}		1.00×10^{-4}
Sb-122	1.53×10^{-5}	1.88×10^{-5}	3.00×10^{-5}
W-187	2.64×10^{-4}		7.00×10^{-5}
Cr-51		2.62×10^{-5}	2.00×10^{-3}
Ag-110m		6.68×10^{-6}	2.00×10^{-4}
H-3	5.00×10^{-3}		3.00×10^{-3}

There was every reason to believe that the water seeped into the gravel bed around the foundation of the reactor building.

Corrective Action

The line from the cold sump to the tile drains was permanently sealed.

The cold sump was returned to its normal configuration. Water is pumped from the sump to the facility's retention tanks before the level reaches any of the collection drain pipes and the sealed pipe to the drain tiles.

Normal leakage collection lines have been diverted to the hot sump. In order for reactor pool water to reach the cold sump, the hot sump pump would have to fail, the hot sump would have to fill, and water would have to flow through the overflow to the cold sump.

The leaking check valve in the cold sump has been replaced. The replacement valve had no back-leakage when tested.

Small sampling wells were drilled at five locations around the reactor building to sample and verify the extent of ground water contamination. Tritium levels below maximum permissible concentrations to the unrestricted area were detected in the well closest to the release point. No other activity above background was detected in any other wells. The migration time for the other nuclides involved in the release is quite long, so low levels may be detected in the future.

A pathway analysis was performed. It showed that the dose to humans by direct ingestion of the well water contaminated by the release would be less than 33 mrem. Direct ingestion had the potential for causing the highest human dose. No one actually drank this water so the actual dose was zero.

A preliminary survey of all pipes to and from the reactor building was conducted. No additional, previously unknown release paths from the building were found. A more detailed survey including a review of building drawings is in progress.

2. POWER GENERATION SUMMARY

The following table summarizes reactor power generation for 1993.

Cycle	Inclusive Dates	Operating Hours	Full Power Operating Hours	Megawatt Hours	Percent Availability
350	01/13/93-02/09/93	482.1	457.2	926.2	56
351	02/10/93-03/09/93	481.7	434.4	871.3	72
352	03/10/93-04/06/93	482.8	426.5	863.3	72
353	04/07/93-05/04/93	462.4	279.4	725.6	69
354	05/05/93-06/01/93	447.2	414.4	833.6	67
355	06/02/93-06/29/93	483.4	450.0	905.0	72
356	06/30/93-07/27/93	405.3	358.5	724.2	60
357	07/28/93-08/24/93	467.0	425.2	855.1	70
358	08/25/93-09/27/93	512.5	459.2	922.5	70
359	09/28/93-10/21/93	307.7	29.4	253.3	46
360	10/22/93-11/18/93	470.2	434.7	878.4	70
361	11/19/93-12/15/93	399.0	257.1	641.4	59
362	12/16/93-01/12/94	280.8	253.8	559.6	42
TOTAL		5,682.1	4,679.8	9,959.5	65

3. UNSCHEDULED REACTOR SHUTDOWN SUMMARY

The following table summarizes unscheduled reactor shutdowns for 1993.

3.1 Unscheduled Shutdowns

Total Unscheduled Shutdowns............20

Operating Hours per Shutdown..........284

3.2 Shutdown Types

Single Rod Drop (NAR)...................5

Multiple Rod Drop (NAR).................0

Operator Action.........................1

Operator Error..........................0

Process Equipment.......................0

Reactor Controls.......................10

Electric Power Failure..................4

3.3 Shutdown Type Definitions

Single Rod Drop and Multiple Rod Drop (NAR)

An unscheduled shutdown caused by the release of one or more of the reactor shim-safety rods from its electromagnet, and for which at the time of the rod release, no specific component malfunction and no apparent reason (NAR) can be identified as having caused the release.

Operator Action

A condition exists (usually some minor difficulty with an experiment) for which the operator on duty judges that shutdown of the reactor is required until the difficulty is corrected.

Operator Error

The operator on duty makes a judgement or manipulative error which results in shutdown of the reactor.

Process Equipment

Shutdown caused by a malfunction in the process equipment interlocks of the reactor control system.

Reactor Controls

Shutdown initiated by malfunction of the control and detection equipment directly associated with the reactor safety and control system.

Electrical Power Failure

Shutdown caused by interruption in the reactor facility electric power supply.

3.4 Cycle Summary of Unscheduled Shutdowns

Cycle 352

There were five unscheduled shutdowns during Cycle 352. All were NAR single rod drops. The rod drops resulted from low magnet holding currents (approximately 54 ma maximum from the safety system). An intermediate power supply has been installed that allows magnet currents up to 90 ma. This power supply is the first step in the safety system upgrade

Cycle 354

There was one unscheduled shutdown during Cycle 354. This shutdown was by operator action when during routine operation, the crew discovered a leaking city water pipe. In order to stop the leak, it was necessary to valve off the city water to the reactor building. This in turn required that the reactor be shutdown due to the lack of the water supply for the emergency pool fill. The reactor remained shutdown until the emergency water fill capability was restored. This took about 15 hours. An additional isolation valve was installed in the system which, if the same system leaked again, allows continued availability of emergency fill.

Cycle 356

There were three unscheduled shutdowns during Cycle 356. The first was a reactor scram due to a power flicker. The second and third were due to an apparent ramp of safety channel A. The high voltage connector to the detector was found to be defective and was repaired. There has been no recurrence of the observed channel A noise or intermittent alarms.

Cycle 357

There was one unscheduled shutdown during Cycle 357 due to loss of power. The problem originated in Detroit Edison's equipment. The reactor was restarted without difficulty.

Cycle 358

There was one unscheduled shutdown during Cycle 358. There was an offsite loss of power due to Detroit Edison working nearby. The reactor was restarted without difficulty.

Cycle 360

There was one unscheduled shutdown during Cycle 360. It was due to a power failure on Safety Channel A. There were no indicators as to the cause or source of this trip. Trouble shooting traced the cause to a corroded connector on the Channel A ion chamber.

Cycle 361

There were three unscheduled shutdowns during Cycle 361. The first was due to loss of electrical power to the building. The loss of power was attributed to equipment outside of the Phoenix Lab. The other two shutdowns were due to reactor scrams on Safety Channel A with no indication of high power. Troubleshooting efforts have not yet revealed the root cause. All connectors were replaced on A Channel and the ion chamber was satisfactory when inspected and tested.

Cycle 362

There were five unscheduled shutdowns during Cycle 362. These unscheduled shutdowns were due to intermittent noise spikes in the ion chamber signal or to level channel electronics. Troubleshooting efforts are ongoing. The intermittent nature of the problem prevents easy analysis. A sophisticated digital oscilloscope has been purchased to aid troubleshooting. There is no indication of high power levels or short periods associated with the shutdowns.

4. CORRECTIVE MAINTENANCE ON SAFETY RELATED SYSTEMS AND COMPONENTS

None

5. CHANGES, TESTS, AND EXPERIMENTS CARRIED OUT WITHOUT PRIOR NRC APPROVAL PURSUANT TO 10CFR50.59(a)

5.5 <u>Modification Request 110 - Replace Underwater Shim-Safety Rod Magnets with Above Surface Magnets</u>

Above surface magnets have been installed on the three shim-safety rods. They were directly compatible with the old magnet power supply and the replacement power supply that was part of the safety system upgrade. The entire rod and magnet holddown system has been tested for operation and rod drop times prior to installation. A final preinstallation test of the final design was conducted with a dummy fuel element and the actual design.

Installation of the above surface magnets was completed during the week of September 18 - 25, 1993. The reactor was kept shutdown during that period, except for necessary operation to test the installation, in order to train the operators on the new system.

5.6 Modification Request 113A - Replace Safety Channels A and B and Log N with One-For-One State of the Art Channels

The upgraded analog safety channels are one-for-one replacements for the old channels. They use the same cables and connectors. The units have been designed and built by General Atomics and have been installed and tested in the General Atomics TRIGA reactor facility. They are also installed at other reactors around the country.

The facility electronics engineer reviewed the installation of these units at General Atomics and at McClellan Air Force Base, and he performed field tests of the completed and interwired units at General Atomics in San Diego before delivery.

Bench tests were completed. The interface between the old and upgraded systems was installed on September 7, 1993. It was QA checked and operationally tested on the bench, and was QA tested again after actual installation. New safety channels A and B were preliminarily tested, one at a time, at the same time.

Final installation of channels A and B and the Log N (channel C) was completed during the week of October 2 - 8, 1993. The reactor remained shutdown for that week in order to complete the installation and to familiarize and train the operators on the new system.

6. RADIOACTIVE EFFLUENT RELEASE

Quantities and types of radioactive effluent releases, environmental monitoring locations and data, and occupational personnel radiation exposures are provided in this section.

6.1 Gaseous Effluents - ^{41}Ar Releases

Gaseous effluent concentrations are averaged over a period of one year.

	Quantity	Unit
a. Total gross radioactivity.	39.3	Ci
b. Average concentration released.	1.15×10^{-7}	µCi/ml
c. Average release rate.	1.24	µCi/sec
d. Maximum instantaneous concentration during special operations, tests, and experiments.	Not Applicable	µCi/ml
e. Percent of ^{41}Ar MPC (4.0×10^{-8} µCi/ml) without dilution factor.	287	Percent
f. Percent of ^{41}Ar MPC with 400 dilution factor.	0.72	Percent

6.2 Radiohalogen Releases

a. Total iodine radioactivity by nuclide based upon a representative isotopic analysis. (Required if iodine is identified in primary coolant samples or if fueled experiments are conducted at the facility). The analysis is based on primary coolant activity following one week of decay.

Iodine-131 was identified in the one week count of the primary coolant samples four times. The concentrations of these occurrences are shown.

I-131 concentration

		Quantity	Unit
1.	02/04/93	5.0×10^{-7}	µCi/ml
	03/04/93	6.9×10^{-7}	µCi/ml
	05/11/93	1.0×10^{-6}	µCi/ml
	07/01/93	9.0×10^{-7}	µCi/ml

Based on the uncertainty of measurements at this concentration, these 4 positive measurements (out of the 52 total measurements taken) are consistent with normal statistical variations. These individual measurements do not indicate any evidence of a new trend. Finally, the magnitudes of the values are lower than those found in a previous fuel element leak.

Xenon-133, a sensitive fission product indicator, was not identified in the one week count of the primary coolant in 1993.

None of these concentrations were indicative of leaking fuel.

b. ^{131}Iodine releases related to steady state reactor operation (Sample C-3, main reactor exhaust stack).

	Quantity	Unit
1. Total iodine released.	207	µCi
2. Average concentration released.	1.0×10^{-12}	µCi/ml
3. Percent of ^{131}I MPC (1.0×10^{-10} µCi/ml) without dilution factor.	1.0	Percent
4. Percent of ^{131}I MPC with 400 dilution factor.	0.003	Percent

c. Radiohalogen releases related to combined steady state reactor operation and radiation laboratory activities (Sample C-2; combined secondary reactor exhaust and partial radiation laboratory exhaust).

	Quantity	Unit

1. Total C-2 stack radiohalogen releases

	Quantity	Unit
Br-82	92,922	µCi
I-123	408	µCi
I-125	340	µCi
I-131	1,270	µCi
I-133	688	µCi

2. Average concentration release.

	Quantity	Unit
Br-82	$6.6x10^{-10}$	µCi/ml
1-123	$2.9x10^{-12}$	µCi/ml
I-125	$2.4x10^{-12}$	µCi/ml
I-131	$9.0x10^{-12}$	µCi/ml
I-133	$4.9x10^{-12}$	µCi/ml

3. Percent of MPC without the dilution factor.

	Quantity	Unit
Br-82	1.7	Percent
I-123	2.9	Percent
I-125	3.0	Percent
I-131	9.0	Percent
I-133	1.2	Percent

4. Percent of MPC with factor of 400 dilution factor.

	Quantity	Unit
Br-82	0.004	Percent
I-123	0.007	Percent
I-125	0.008	Percent
I-131	0.023	Percent
I-133	0.003	Percent

d. Total Facility Release of Radiohalogens

1. Total facility radiohalogen releases

	Quantity	Unit
Br-82	113,772	µCi
I-123	232,200	µCi
I-125	11,529	µCi
I-131	53,298	µCi
I-133	3,424	µCi

	Quantity	Unit
2. Average concentration released		
Br-82	1.9×10^{-10}	μCi/ml
I-123	3.8×10^{-10}	μCi/ml
I-125	1.7×10^{-11}	μCi/ml
I-131	7.8×10^{-11}	μCi/ml
I-133	5.6×10^{-12}	μCi/ml
3. Percent of MPC without dilution		
Br-82	0.46	Percent
I-123	377	Percent
I-125	21	Percent
I-131	78	Percent
I-133	1.4	Percent
TOTAL	478	Percent
4. Percent of diluted MPC using factor of 400 dilution		
Br-82	0.001	Percent
I-123	0.94	Percent
I-125	0.052	Percent
I-131	0.19	Percent
I-133	0.0034	Percent
TOTAL	1.2	Percent

6.3 Particulate Releases

Gross alpha activity is required to be measured if the operational or experimental program could result in the release of alpha emitters.

a. Total gross beta-gamma radioactivity.	139	μCi
b. Gross alpha radioactivity.	Not Required	
c. Total gross radioactivity of nuclides with half lives greater than eight days.	2.1×10^{-13}	μCi/ml
d. Percent of MPC (1.0×10^{-10} μCi/ml) for particulate radioactivity with half lives greater than eight days without dilution factor.	0.21	Percent

	Quantity	Unit
e. Percent of MPC for particulate radioactivity with half lives greater than eight days with 400 dilution factor.	0.0005	Percent

6.4 **Liquid Effluents**

No radioactive effluents were released from the facility in 1992.

6.5 **Environmental Monitoring**

The environmental monitoring program for the Ford Nuclear Reactor facility consists of direct radiation monitors (TLD) and air sampling stations located around the facility and selected water and sewer sampling stations.

a. **TLD Environmental Monitors**

TLDs located at stations to the north (Northwood Apartments, Automotive Laboratory, and lawn adjacent to the reactor building), east (Industrial and Operations Engineering), south (Glazier Way and Institute of Science and Technology), and west (Chrysler Center and School of Music) of the reactor facility are collected and sent to a commercial dosimetry company for analysis.

Station Description	Annual Dose	Unit
Northwood (N)	74	mrem
Lawn (N)	104	mrem
Ind. and Operations Eng. (E)	73	mrem
Glazier Way (S)	57	mrem
Institute of Science and Technology (S)	88	mrem
Chrysler Center (W)	66	mrem
Automotive Laboratory (N)	88	mrem
Distance #1	103	mrem
Distance #2	64	mrem
Control (Stored in lead pig)	30	mrem

Background is taken at a distance in excess of two miles from the reactor (Distance 1 and Distance 2). As none of the indicator badges were distinguishable from background, the average dose is reported as zero (0).

b. **Dust Samples**

Four air grab samples are collected from continuously operating monitors located to the north (Northwood Apartments), east (Industrial and Operations Engineering), south (Institute of Science and Technology), and west (Chrysler Center) of the reactor facility. Each filter sample is counted for net beta activity. The unweighted mean radioactivity concentrations are shown below.

Station Description	Mean Concentration	Unit
Northwood (N)	1.0×10^{-14}	µCi/ml
Ind. and Operations Eng. (E)	2.5×10^{-14}	µCi/ml
Chrysler Center (W)	6.3×10^{-15}	µCi/ml
Institute of Science and Technology (S)	1.1×10^{-14}	µCi/ml

The result of the air sampling expressed in percentages of maximum permissible air concentrations are shown below:

Station Description	Mean MPC Value	Unit
Northwood (N)	0.10	Percent
Industrial and Operations Engineering (E)	0.25	Percent
Chrysler Center (W)	0.06	Percent
Institute of Science and Technology (S)	0.11	Percent

No significant environmental particulate emmissions are expected since the stack particulate releases are negligable. None of these locations show a statisically significant increase from each other (95 confidence level).

c. **Water Samples**

Since the facility does not release any liquid radioactive effluents, the water sample data is not applicable and is not included.

d. **Sewage Samples**

Since the facility does not release any liquid radioactive effluents, the sewage sample data is not applicable and is not included.

e. **Maximum Cumulative Radiation Dose**

The maximum cumulative radiation dose which could have been received by an individual continuously present in an unrestricted area during reactor operations from direct radiation exposure, exposure to gaseous effluents, and exposure to liquid effluents:

1. Direct radiation exposure to such an individual is negligible since a survey of accessible areas around the reactor building shows no detectable radiation dose rates above background.

2. Gaseous Effluents

   The gaseous effluents from the reactor and the contiguous laboratory facility are as follows:

Isotope	Total Release (μCi)	Concentrat. (μCi/ml)	Percent of MPCu Undiluted	Percent of MPCu Diluted
^{41}Ar	3.93×10^{7}	1.15×10^{-7}	287	0.7200
^{82}Br	113,772	1.85×10^{-10}	0.46	0.0012
^{123}I	232,200	3.77×10^{-10}	377.25	0.9430
^{125}I	11,529	1.68×10^{-11}	20.96	0.0500
^{131}I	53,298	7.75×10^{-11}	77.53	0.1940
^{133}I	3,424	5.56×10^{-13}	1.39	0.0035
Gross Particulate	139	2.06×10^{-13}	0.21	0.0005
		TOTAL	764.80	1.9122

   The total gaseous effluent releases are well within the allowed release concentrations when the conservative dilution factor of 400 is applied.

3. Liquid Effluents

   The annual dose from liquid effluents is zero since this facility does not release any liquid radioactive effluents.

f. If levels of radioactive materials in environmental media, as determined by an environmental monitoring program, indicate the likelihood of public intake in excess of 1% of those that could result from continuous exposure to the concentration values listed in Appendix B, Table II, 10CFR20, estimate the likely resultant exposure to individuals and to population groups and the assumptions upon which those estimates are based.

Exposure of the general public to 1 MPCu would result in a whole body dose of 500 mrem. The maximum public dose based on gaseous effluent releases of 1.91 % MPCu is 9.6 mrem. This dose is based on a member of the public continuously breathing airborne radioactivity at the point of minimum dilution near the reactor building.

6.6 Occupational Personnel Radiation Exposures

Two hundred and fifty two facility operational personnel were provided personal monitors. Individuals for whom extremity monitoring was provided received TLD ring dosimeters for each hand. No radiation exposures greater than 50 mrem were received at the facility by individuals under the age of 18. There were no declared pregnant females at the facility.

A summary of whole body exposures based upon data from January 1 through December 31, 1993 is as follows:

Estimated Whole Body Exposure Range (rem)	Number of Individuals in Each Range
No measurable exposure	188
Measurable exposure, less than 0.10	40
0.10 - 0.25	7
0.25 - 0.50	8
0.50 - 0.75	4
0.75 - 1.00	2
1.00 - 1.25	2
1.25 - 1.50	1
Greater than 1.50	0
Total	252

Maximum individual whole body exposure: 1.27 rem
Cumulative "deep" whole body exposure: 12.79 rem
Mean "deep" whole body exposure: 51 mrem