H.C. 3.01; MN 625

MONTHLY REPORT NO. 4 - NOVEMBER 1966

COMPILATION OF CURRENT TECHNICAL EXPERIENCE AT
ENRICO FERMI ATOMIC POWER PLANTAEC CONTRACT NO. AT (11-1) -865
PROJECT AGREEMENT NO. 15SPONSOR: DIVISION OF REACTOR DEVELOPMENT
AND TECHNOLOGY, UNITED STATES
ATOMIC ENERGY COMMISSIONPrepared by
ATOMIC POWER DEVELOPMENT ASSOCIATES, INC.
DETROIT, MICHIGAN
with the cooperation of
POWER REACTOR DEVELOPMENT COMPANY
AND
THE DETROIT EDISON COMPANYApproved Charles E. Branyan
Charles E. Branyan
Fermi Project Engineer
APDAAuthor: J. F. McCarthy
Commonwealth Associates Inc.
Jackson, Michigan

Date Issued: May 1967

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission.

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

CONTENTS

<u>Section</u>	<u>Page</u>
PREFACE	3
I CURRENT EXPERIENCE SUMMARY	5
II DATA AND STATISTICS	7
A. Plant Operations	7
1. Reactor Operating Statistics	7
2. Daily Operation Curve Sheets	8
3. Reactor Loading	12
4. Fuel Handling Operations	14
5. Sodium and Gas Systems Performance	15
B. Plant Repairs and Modifications	17
1. Primary Sodium Pumps	17
2. Secondary Sodium Pumps	18
3. Reactor Safety Systems	18
4. Fuel Handling Equipment	18
5. Steam Generators	20
6. Raising of the Reactor Hold-Down Mechanism	20
7. Operation of the Reactor	26
8. Core Investigation Using the OHM	26
III NUCLEAR TEST PROGRAM	31
IV PLANT TEST PROGRAM	31
V OTHER OPERATING EXPERIENCE	32
A. Steam Generator Stability	32
1. General	32
2. Observed Instabilities	32
3. Explanation	33
4. Thermal Inertia	33
5. Future Operating Conditions	35
VI MAINTENANCE AND ADDITIONS	36
A. Thawing the Reactor Vessel Syphon Drain Line	36

PREFACE

PURPOSE

The purpose of this monthly report is to make available to the fast reactor program the current experience being gained from the Enrico Fermi Atomic Power Plant.

SCOPE

The scope of this report includes all phases of current nuclear operating and maintenance experience at the Enrico Fermi Atomic Power Plant.

Earlier Fermi experience in certain selected areas is being recorded in a series of technical reports completed or in preparation by the Atomic Power Development Associates Inc. for the US Atomic Energy Commission under AEC Contract No. AT(11-1)-865, Project Agreement No. 15. This series of reports provides detailed information on the nuclear testing, rod drives, machinery dome, fuel subassemblies, fuel handling, steam generators, pumps, flowmeters, level detectors, sodium sampling and development of the primary sodium system.

The items in Section II are usually reported each month; items in the other sections are selected on the basis of their special significance during the month. Other items may be found in the monthly report submitted to the Atomic Energy Commission by Power Reactor Development Company in compliance with the requirements of Provisional Operating License No. DPR-9, as amended.

BACKGROUND

The Fermi reactor achieved initial criticality on August 23, 1963. An extensive series of nuclear tests was conducted at power levels below one megawatt thermal, through 1965. A high power (200 Mwt) license was issued on December 17, 1965, and operation in excess of 1 Mwt was initiated on December 29, 1965. In January 1966, the power was raised in a series of steps to 20 Mwt. On April 1, 1966, power was first raised to 67 Mwt and on July 8, 1966, operation at 100 Mwt was initiated.

The scope of the first several monthly reports will not be limited to items occurring since August 1, but will also include several earlier items of experience dating back as far as April 1, 1966.

It is assumed that those reading this report have a general familiarity with the plant. As an aid to the reader, a perspective drawing of the plant was included at the back of report No. 1.

Note

Since this report is intended to follow closely the current proceedings at the Fermi plant, it must necessarily be treated as preliminary information, subject to supersedence in the light of subsequent experience.

SECTION I

CURRENT EXPERIENCE SUMMARY

The reactor remained shutdown because of the fuel damage sustained on October 5, 1966. Early in the month the hold-down mechanism was raised, disengaging the heads of the core subassemblies. This operation was completed without difficulty, following a carefully prepared detailed plan.

The reactor sweep mechanism was used to establish that no subassemblies are stuck to the hold-down mechanism (HDM) and that no large pieces of debris are lying on top of the reactor subassemblies.

The offset handling mechanism (OHM) was used to verify the position of the subassemblies and to measure the force required to start lifting them.

The first probing with the OHM was done in the area of subassemblies M140 and M098 which had shown high sodium outlet temperatures. Forty core, inner radial blanket (IRB) and adjacent outer radial blanket (ORB) subassemblies were successfully checked for position, then 17 of these were lifted less than 3/4-inch and reinserted. Subassembly M098 offered unusual resistance so it was left 300 mils high.

Next, the OHM was used to verify the position of the handling heads of the remainder of the core and inner radial blanket subassemblies and of fifteen additional outer radial blanket subassemblies adjacent to the IRB subassemblies. All the subassembly heads were located by the OHM. One subassembly was 0.140-inch low, another 0.075-inch high, but most were within 0.050-inch of the expected elevation which is within the normal accuracy of the OHM calibration.

Lifting of the subassemblies with the OHM was then resumed, this time with a limit of approximately 350 lb force. Nine subassemblies would not lift under this restriction. The lifting force limit was increased to 550 lb and three subassemblies still would not lift.

See page 26 for details on the use of the OHM for probing the core.

Subassembly M099, which had been removed from the core in July, (see Page 11 of Report No. 2) was transferred by the offset handling mechanism from an outer storage position in the reactor to the reactor transfer rotor. Later, it was transported by the cask car to the transfer tank in the Fuel and Repair Building. It is scheduled to be cleaned, cut into sections and shipped to the Battelle Memorial Institute hot lab for destructive examination.

SECTION I

It was decided to proceed with shipping cask modifications which will permit full length core subassemblies to be shipped to Battelle Memorial Institute for inspection. Detailed design is commencing on a fuel examination facility to be located at the reactor exit port in the containment building. More information on these items will be reported in the December monthly report when these programs will be more fully defined.

SECTION II
DATA AND STATISTICS

A. Plant Operations

1. <u>Reactor Operating Statistics</u>	<u>November</u>	<u>Cumulative through November</u>
Reactor Start-ups	-	406
Scheduled Scrams (Includes usual test scram prior to reactor start-up)	-	718
Unscheduled Scrams	-	56
Maximum Reactor Power, Mwt	-	114*
Integrated Reactor Power Mwt - days	-	855*
Hours of Reactor Operation	-	2089
Maximum Burn-up in Central Core Subassembly, a/o	-	0.06
Maximum Electrical Output, Gross Mwe	-	22
Integrated Electrical Output, Gross Kwhr	-	1,160,000

* Indicated values - actual values are 10 percent lower, based on heat balance data - Further data to be taken and analyzed before final corrected values are established.

SECTION II
DATA AND STATISTICS

A. Plant Operations

2. Daily Operation Curve Sheets

See the following three fold-out pages. Since there was no reactor operation during November the data shown are for July 1966. July was a representative month, with reactor operation at both 67 Mwt and 100 Mwt. See pages 1 and 3 of Report No. 1 for related information on July operation.

The primary sodium flows are indicated values - actual values are approximately 10 percent lower based on heat balance data. Further data will be taken and analyzed before final corrected values are established.

FERMI DAILY REACTOR PLANT OPERATION DURING THE MONTH OF JULY 1966

Loop No. 1	Pump Running Time: 742	Hours This Month, 14,976	Hours Cumulative
Loop No. 2	404	12,162	
Loop No. 3	671	18,172

Time Remaining: 2 sec This Month: 26, 12 / Hours Cumulative

Pump running Time: 614 Hours This Month, 44,529 Hours Cumulative

Pump Running Time: 646 Hours This Month 23,934 Hours Cumulative

FERMI DAILY REACTOR PLANT OPERATION DURING THE MONTH OF JULY 1966

Notes: Values are Average of All Loops in Service

FERMI DAILY REACTOR PLANT OPERATION DURING THE MONTH OF JULY 1966

SECTION II
DATA AND STATISTICS

A. Plant Operation

3. Reactor Loading

The diagram on the following page shows the present reactor loading. Subassembly M099 was removed from a storage position as described on Page 14. The diagram emphasizes the location of reactor subassemblies M140 (NO4-PO4) and M098 (NO3-PO3) which had high sodium outlet temperatures during the October 5th operation and now show high lifting forces.

N16 N15 N14 N13 N12 N11 N10 N09 N08 N07 N06 N05 N04 N03 N02 N01 N00 P01 P02 P03 P04 P05 P06 P07 P08 P09 P10 P11 P12 P13 P14 P15 P16

KEY:

CR Control Rod No. 430169-
 SR Safety Rod No. 430192-
 OR Oscillator Rod
 CS Core Shim Subassembly
 CF Core Foil Subassembly
 BF Blanket Foil Subassembly
 CT Coarse Filter, Take-apart, Dummy Core Subassembly
 NOTE: Dummy Core subassemblies in the reactor meet "Core A" core subassembly specifications and bear the suffix "CF"

NA Sodium Worth Subassembly
 NS Neutron Source
 TIT Temporary Instrument Thimble
 MS APDA Materials Surveillance Subassembly
 M Subassembly Manufactured by D.E. Makepeace Co.
 S Subassembly Manufactured by Sylvania Division, Sylvania Electric Products Co.

M 001 - M 206 Core Subassemblies
 M 301 - M 400 Inner Radial Blanket Subassemblies
 S 500 - S 798 Outer Radial Blanket Subassemblies
 M 801 - M 1000 Outer Radial Blanket Subassemblies

Units shown without prefix are dummy outer radial blanket subassemblies.

- Oversize Nozzle Unit
- "F" Subassembly (Contains fuel pins with high iron plus nickel, high carbon or Zirconium content.)
- "W" Subassembly (Contains fuel pins with high iron plus nickel content.)
- Blanket slugs have high carbon content. (APDA Surveillance Program Unit)
- Stringering in Blanket Slugs
- Large Grain Blanket Material (Hash)
- Larger Than Normal Spacing Between the Blanket Elements and the Support Grid
- Type 347 Stainless Steel Wrapper Tube
- Handling Head Short
- Test Flow Subassembly (S-400)
- Slugs Previously Used in a Test Subassembly
- "CP" Slugs
- Space Where Subassembly Was Removed

REACTOR LOADING

SECTION II
DATA AND STATISTICS

A. Plant Operations

4. Fuel Handling Operations

a. Transfer of Subassembly MO99

Before commencing 100 Mwt reactor operation in July 1966, one central core subassembly, MO99, was removed from core position NO1-PO1 and a new subassembly inserted in its place. The MO99 subassembly had been placed in an outer storage position in the reactor for possible future examination.

The subassembly was transferred from the storage position to the reactor transfer rotor in early November soon after the hold-down mechanism was raised. The offset handling mechanism was manually operated during this transfer. It was then carried by the cask car to the transfer tank in the Fuel and Repair Building (see page 18). It will be steam cleaned, cut into sections and sent to the hot laboratory at Battelle Memorial Institute for destructive examination. This subassembly may provide evidence as to the cause of the October 5th fuel damage and will provide a standard of comparison when the faulty subassemblies are examined at BMI.

b. Steam Cleaning Machine

Acceptance testing of the steam cleaning machine in the Fuel and Repair Building was started in October and continued throughout November. See Section II, A4, of the October experience report for further details on this testing, and page 19 of this report for recent steam cleaning chamber gripper modifications. A total of eight dummy fuel subassemblies were cleaned during November, with five remaining to be cleaned to achieve the desired total of twenty.

The machine was successfully tested to demonstrate proper functioning of the control system which takes over automatic operation of the machine under emergency conditions. Scram action would be initiated by the operator if an emergency forced him to leave the Fuel and Repair Building. If the scram occurs before the cleaning chamber lower valve has closed, the subassembly is returned to the transfer tank; otherwise the cleaning is completed and the subassembly automatically lowered into the cut-up pool water.

SECTION II A

c. Fuel Handling Statistics

	Total Movements 9/28/66 to 11/28/66	Cumulative Totals As of 11/28/66
Reactor Hold- down Mechanism *	1	539
Reactor Rotating Plug	250 estmd	5,400
Reactor Offset Handling Mech.		
Azimuth	250 estmd	7,600
Elevation	500 estmd	8,500
Cask Car	3 **	477
Steam Cleaning Machine - Cleaning Operations	15	25
Fuel Cut-Up Machine - Subassemblies Sectioned	1	Approx 11

* One count per complete raise and lower cycle.

** 9/1/66 to 11/28/66, Round trips.

5. Sodium and Gas Systems Performance

a. Sodium Cold Traps and Plugging Indicators

November Operating Data	Primary System	Secondary System		
		Loop 1	Loop 2	Loop 3
Cold Trap Operation (Hours)	8	0	0	0
Maximum Plugging Temperature - F	below 220	260	310	250
Minimum Plugging Temperature - F	below 220	240	275	210

b. Primary System Cover Gas Analysis

	Reactor Cover Gas (Argon) ppm by Volume	Primary Shield Tank Atmosphere (Nitrogen) ppm by Volume
Oxygen	below 25	45*
Carbon Monoxide	below 10	below 10
Carbon Dioxide	10	35
Hydrogen	below 4**	2.5
Helium	below 4	below 4
Methane	below 10	below 10
N ₂ O	not measured	below 10
NO ₂	not measured	below 1
Argon	remainder	not measured
Nitrogen	1300	remainder
Analysis Date	Nov 18	Nov 18

* Technical Specifications state 1000 ppm maximum.

** 10 ppm recommended maximum.

c. Primary Sodium Chemical Analysis

Sample Date	Oct 15	Nov 20
Total Oxygen	9,9,11	11,11,12
Total Carbon	48,51,177,200	43,46,55,84
*Nonhydroxide Hydrogen	0.3,0.3	0.4
*Hydroxide Hydrogen	1.4, 1.4	1.4
Iron	0.1, 0.2	0.7, 0.9
Nickel	below 0.2,0.1	below 0.3, below 0.3
Chromium	below 0.1, below 0.2, below 0.2	0.5, 0.6

Note: Values are in ppm. One sodium sample coil was analyzed at several different points along its length to provide the separate readings indicated.

* 1.3 ppm recommended maximum total hydrogen.

d. Sodium Activity - Maximum Values for November

Primary System 0.087 microcuries/cc Na. Sample
date 11/20. Analysis date 12/27

Secondary System Below instrument minimum
sensitivity of 2.5×10^{-4}
microcuries Na-24/cc Na.

e. Primary Cover Gas Activity

<u>Location</u>	<u>Sample Date</u>	<u>Concentration (microcuries/cc)</u>
Reactor Cover Gas	11/8/66	1.1×10^{-3}
Primary Shield Tank	11/11/66	6.2×10^{-7}
Pump Tank No. 1	11/18/66	1.4×10^{-4}

SECTION II

DATA AND STATISTICS

B. Plant Repairs and Modifications

1. Primary Sodium Pumps

a. The main motors on the primary sodium pumps were shut off before raising the hold-down mechanism on November 1. The usual procedure is to reduce the pump speed to nominal refueling flow, which is about one-half rated flow, before raising the HDM. However, because of the uncertainty regarding the condition of the core subassemblies after the fuel damage on October 5, 1966, the primary pumps were switched to pony motor operation before raising the HDM on November 1. On pony motor operation, the flow is approximately 0.5×10^6 lb/hr per loop, or 15 percent of rated flow.

b. The pony motors impart considerably less energy to the main primary sodium system, so some of the electrical heaters have been turned on to help maintain the primary system temperature at around 480 F. Forty-five of the 108 main primary sodium heating circuits are at present energized. In addition some heat is being transferred into the primary system via the intermediate heat exchangers, from the secondary sodium system. The three secondary pumps are being operated at or near normal speeds.

When the offset handling mechanism is in use (see page 26) normal operating procedures require that the level of the sodium in the reactor vessel be lowered slightly so that the OHM bellows will not be immersed in sodium during fuel handling operations. There is some uncertainty as to whether this lower sodium level provides sufficient head for proper operation of the main pump bearings during pony motor operation when the pump speed is reduced. Therefore, the pony motors operate the main pumps for only about 5 minutes each hour when the sodium is at the lower level.

c. During the month, the cask car was brought into the reactor building to remove subassembly M099 from the reactor transfer rotor container, as described on page 18. The plant Technical Specifications require that "containment building integrity must be maintained whenever the reactor is not secured." The specifications also stipulated that the hold-down mechanism be in the "down" position to meet one of the requirements for securing the reactor. With these requirements, the building equipment door could not be opened to admit the cask car unless the HDM were relowered. Lowering the HDM at this time was undesirable because some of the subassembly heads may have shifted as a result of the October 5, 1966, fuel melting and hold-down finger damage could occur. Therefore, a request, with supporting data,

SECTION II B

was made to the AEC Division of Reactor Licensing to change the requirement on securing the reactor to stipulate that the hold-down mechanism be in the "down" position or that the power supply to each of the three main primary sodium pumps has been physically disconnected by delatching the electric cables from the circuit breakers. This request was granted on November 21, 1966. The cables were removed, permitting the building door to be opened with the HDM in the raised position and the transfer of MO99 was accomplished.

2. Secondary Sodium Pumps

There were no significant items of maintenance on the secondary sodium pumps during November.

3. Reactor Safety Systems

A daily check is made of the source range count rate, using the high sensitivity neutron detector rack No. 4 in parallel with source range rack No. 3.

The reactor safety system has been modified by the addition of load resistors to the output of the solid state relays. The resistors keep a light load on the relays when the key interlock circuit to the safety rods is open. This results in more comparable operation of the relays and prevents false comparator alarms when handling fuel.

4. Fuel Handling Equipment

a. Cask Car

An overall leak check of the cask car showed the argon out-leakage to be 10 cubic feet per hour at a pressure of 8-inches wcg. Investigation revealed several faulty welds, which were repaired, and faulty seals which were replaced. A second leak test gave an acceptable out-leakage rate of 8 cubic feet per hour at 8-inches wcg.

The cask car was subsequently moved into the reactor building, bringing one drainable finned pot and one normal (nondrainable) finned pot for carrying subassemblies to the Fuel and Repair Building (FARB). Subassembly MO99 was transported to the sodium filled transfer tank in the FARB in the drainable pot. This type of pot was employed to minimize the carry-over of radioactive sodium. The decay heat in the subassembly was sufficiently low (less than 100 w) that it was unnecessary to have sodium in the pot or forced cooling.

Special precautions were taken in case sodium oxides containing fission products were to become air-borne. The car was thoroughly heated and then, to reduce chances of internal spillage or out-leakage, the argon blowers and heaters were turned off and the argon pressure reduced.

SECTION II B

The air in the reactor building was sampled continuously and respirators were available to the operators if needed. A temporary duct was connected to the building exhaust system and extended to the exit port area. A special air sample, taken 10 minutes before the cask car was to leave the exit port, showed the activity to be only 1/200th of the maximum permissible concentration. After the cask car departed, there were only three tiny sodium particles found atop the exit port plug. The largest weighed 0.136 gram and had a combined beta and gamma activity of 0.01 microcuries/gram. The gamma spectrum of the sodium particles agreed very closely with that of Cs-137 and Na-22.

b. Steam Cleaning Machine

After cleaning seven dummy subassemblies in October, the cleaning machine gripper failed to engage the handling head of the next subassembly. On this particular operation the argon blower was "on" as the empty gripper was lowered out of the cleaning chamber and into the sodium-filled transfer tank to lift out the next subassembly. It is not necessary for the blower to be energized and on the previous cleaning cycles it had not been energized. The blower was energized at this time to permit an accurate determination of the weight of the empty gripper.

The gripper was removed from the system for examination and it was concluded that a reaction from the gas flow caused difficulty with an easily movable internal part of the gripper, the bucket. The bucket prevents gripper latching unless located over a finned pot. The reaction of the blower gas caused the bucket to drag excessively on a positioning cam, preventing its rotation which in turn prevented the gripper fingers from latching onto the subassembly.

The gripper was modified by adding a stop to limit the bucket travel so that it does not touch the positioning cam. The gripper now engages the subassembly handling heads satisfactorily when the argon blower is energized.

c. Cut-Up Machine

Preoperational testing and repairs on the fuel subassembly cut-up machine were accelerated during November. The machine will be needed in early December to cut up subassembly MO99 so that it can be fitted into the shipping cask and sent to Battelle Memorial Institute for inspection.

The machine makes two simultaneous transverse milling cuts through the subassembly. This separates the upper and lower axial blanket sections from the center fuel section. Overall length of the three pieces is roughly the same. One end of the center section is crimped to prevent the fuel pin bundle from sliding out of the wrapper can. See pages 21 and 22 for photographs of the machine.

The machine performs the cutting operation under thirty feet of water at the bottom of the cut-up pool. The machine was removed from the pool several months ago for testing and repairs. As a final test the machine segmented an empty dummy subassembly. It was then lowered into the pool and continued to operate without difficulty. As a last check before cutting up subassembly MO99 a dummy subassembly containing internals similar to those in a real subassembly will be cut underwater. Underwater operations are monitored by underwater TV cameras.

The cutter motors and the chip filter pump motor are electric. The cutter head traverse mechanism and the subassembly tilt, grip and crimp mechanisms are hydraulically operated. The working fluid is 600 psi demineralized water in a closed circuit.

5. Steam Generators

The three steam generators were boiled dry by slowing venting to the atmosphere and they were maintained in a dry condition throughout November. The tubes and tube sheets are blanketed with nitrogen at 20 psig to prevent corrosion or in-leakage. Sodium circulation continues in the three secondary loops and shell sides of the steam generators. This stand-by condition will be preserved until the reactor repairs are completed and power operation resumed.

6. Raising of the Reactor Hold-Down Mechanism

The principal function of the hold-down mechanism is to prevent lifting of the subassemblies by the sodium flow up through the core. See page 23 for a view of the HDM. Fingers mounted on the underside of the HDM push down on the tops of the core and inner radial blanket subassemblies and slightly compress a spring at the inlet nozzle of the subassembly in the support plates.

Because of the uncertainty regarding the condition of the core following the fuel damage on October 5, 1966, it was thought possible that one or more subassemblies could be adhering to the hold-down fingers. Hence caution was exercised as the HDM was lifted, since it was conceivable that the core could be disturbed in such a manner as to increase the reactivity. It was necessary to raise the hold-down mechanism in order to expose the top of the core to permit subassembly movement and examination.

The hold-down was raised the first 1/4-inch by manual operation of the drive. Strain gauges were attached to the actuator shafts and the readings recorded. Dial indicators

LOWERING CUT-UP MACHINE INTO POOL

CUT-UP MACHINE CUTTER AND ACCESSORY PIECES

REACTOR VESSEL AND PRIMARY SHIELD TANK

SECTION II B

were attached to the control rod, safety rod and oscillator rod extensions to check that the rod guide tubes were not being lifted up against the rod extensions. The source range neutron detectors were also monitored for possible indications of a change in the core reactivity, and the HDM was acoustically monitored for any unusual sounds. Three pickups were used for this, two recording and one audible.

The HDM was raised in small increments. At the full-down position, the total load on the three actuator shafts was 77,000 lb compression. When the hold-down mechanism had been raised 3/16-inch, the load was 35,000 lb tension. The control, safety and oscillator rod guide tubes appeared to separate without difficulty. There was no significant change in the source range count rate during the raising of the HDM.

The hold-down finger, core inlet and core outlet thermocouples were also observed during the raise operation; no change was noted. In addition, the fission product detector was in service during the HDM raising; nothing unusual was detected.

After the HDM had been raised 3/16-inch there was a delay while the results at that point were reviewed. The indicated lifting force at this point was 2,000 lb higher than anticipated, but the accuracy of the strain gauges could have accounted for this discrepancy. The raising operation was subsequently resumed and the HDM was raised the full 9-1/2-inch without incident. See page 25 for a plot of load versus travel for the first 1/4-inch of lift.

The HDM was raised by its electric motor drive from the 1/4-inch elevation to the 9-inch elevation. The drive was operated remotely from the control room, with no one in the reactor containment building. The Sanborn strain gauge recorders were located outside the reactor machinery dome on the building operating floor. They were monitored from the control room, via television, during this phase of the hold-down mechanism raising operation. The operation was stopped at intervals to check the source range neutron detector count rates and readings of the other instruments.

The HDM was manually raised the last 1/2-inch of travel, to 9-1/2-inch, because the motor drive normally is tripped at 9-inch elevation. To make sure that the HDM could not be inadvertently relowered, blocks were installed between the lower spring plate and the mounting plate. Temporary blocks were first installed, later replaced with pieces of 4-inch pipe.

**LOAD VS TRAVEL DURING SEQUENCE OF EVENTS
AS HOLD-DOWN MECHANISM IS RAISED**

SECTION II B

7. Operation of the Reactor Sweep Mechanism

As soon as the HDM had been raised, the sweep arm was manually operated over the core and inner radial blanket subassemblies. This was done to ensure that no subassemblies had been pulled out of the core with the hold-down column by adhering to the HDM fingers, and that there was no debris lying on top of the subassemblies. A horizontal arm traverses a fan-shaped area and sweeps just above the top of the subassembly handling heads. After the manual sweep of the core area, motorized sweeps were made over the remainder of the reactor and transition deck area. During the sweep operation three sound pickups were used to listen for and record any sounds of the sweep arm encountering an obstacle. No unusual noises or resistance to sweep arm movement were observed.

8. Core Investigation Using the OHM

The offset handling mechanism is used to insert or remove subassemblies or other items into the reactor core or blanket. See page 23 for a view of the OHM.

As a first check, 40 subassemblies in the southwest quadrant of the reactor were engaged by the OHM gripper. The accuracy of locating subassemblies with the OHM is approximately 250 mils laterally and 50 mils vertically. This area contained the two subassemblies with high thermocouple readings, M140 (NO4-PO4) and M098 (NO3-PO3). The check demonstrated that these handling heads had not shifted laterally and were at, or at most 100 mils below, their expected elevation.

The next investigation was to lift 17 subassemblies in the immediate vicinity of M140 and M098. Each of these subassemblies was lifted less than 3/4-inch and then reinserted, indicating that they were not stuck. Subassembly M098 offered considerable resistance, and was left at a plus 300 mils elevation.

The lifting was limited to 3/4-inch because if a subassembly were to become stuck at any higher elevation its protrusion up into the OHM foot would interfere with the rotation of the OHM. See page 28 for a diagram showing the location of the 40 subassemblies which were probed and the 17 which were lifted.

The OHM was then used to probe the remainder (108) of the core and inner radial blanket subassemblies for azimuth and elevation. Fifteen additional outer radial blanket subassemblies immediately outside the inner radial blanket were also probed. The locations and elevations were found to be satisfactory. Most of the elevations were within 50 mils of the zero elevation; however, one subassembly was 140 mils low and one was 75 mils high.

SECTION II B

At this point it was concluded that more accurate information was needed as to the lifting forces required to raise the subassemblies. The as-built loading method of the OHM lent itself well to reasonably accurate determination of the lifting forces. Only minor modifications and additions were necessary. The schematic diagram on page 29 shows the method used to determine the forces. Calibrated lifting and pushing springs were installed. Allowance was made for the buoyancy of the inert gas filled crane tube and the variation of this buoyancy with elevation was minimized by limiting the lift to less than 3/4-inch.

Referring to the diagram on page 29, the OHM gripper is lowered onto the subassembly handling head and raised until all clearances are taken up and the subassembly is ready to lift. Then the ball screws are manually rotated with a torque wrench. This causes the loading plate to rise which, in turn, increases the force which the pulling springs are exerting on the upper overload plate (the pushing springs were usually disengaged at the point at which the subassembly began to lift). The spring travel is measured by the dial indicator. From the travel and spring rate, the approximate lifting force required to raise the subassembly can be calculated. Subassembly motion is determined by observing the upper overload plate motion with respect to a fixed reference. The ball screw torque wrench reading provides an approximate check on the lifting forces required.

The base calibration was accomplished by lifting a core subassembly from a finned pot at the reactor transfer rotor where there are no adjacent subassemblies which could introduce friction forces. All other subassemblies are compared to this base. At the end of November, all core and inner radial subassemblies and 25 ORB subassemblies had been lift-tested by this method. A lifting limit was initially imposed of approximately 350 lb force. This is approximately 250 lb above the weight of a core subassembly, and approximately 200 lb above the weight of an IRB or ORB subassembly. Nine subassemblies would not start to lift when this amount of force was applied.

The lifting force limit was then increased to approximately 550 lb; three subassemblies still would not lift. However, these were among the seventeen which were previously lifted, as described above, with undetermined forces. Possible reasons for this are: (1) subassemblies may have been welded together by high temperatures during the October 5 fuel damage incident; (2) there may be high frictional forces between subassemblies, perhaps due to swelling or warping caused by the October 5 incident; or (3) the subassembly lower nozzles may be binding in the support plate, again perhaps due to the October 5 incident. See page 30 for a diagram of the subassemblies lift-tested to date which require higher than average forces.

CORE INVESTIGATION WITH THE OHM

SCHEMATIC OF OFFSET HANDLING MECHANISM

Investigations Through November 30, 1966

More Than 175 Lb Lifting Force Required

More Than 350 Lb Lifting Force Required

Did Not Lift At 550 Lb Force
Raised With an Undetermined Force > 550 Lbs

CORE LIFT INVESTIGATIONS THROUGH NOVEMBER 30, 1966

SECTION III

NUCLEAR TEST PROGRAM

The nuclear test program has been suspended as a result of the October 5, 1966, fuel damage incident.

SECTION IV

PLANT TEST PROGRAM

The plant test program has been suspended as a result of the October 5, 1966, fuel damage incident.

SECTION V

OTHER OPERATING EXPERIENCE

A. Steam Generator Stability

1. General

Steam generator stability difficulties were experienced in July and August 1966, as noted in Report No. 1 for August 1966. A comprehensive analysis of the problem, with recommendations, was issued at the end of October by Atomic Power Development Associates, and is summarized below.

The following secondary sodium and steam conditions had been predicted (per Volume 2 of the Hazards Summary Report) for 100 Mwt, three-loop operation:

Secondary Sodium Flow Rate = 2.95×10^6 lb/hr/loop

Steam Generator Na Outlet Temp (TE-301) = 517 F

Steam Pressure = 950 psia

Superheated steam would be generated at around 80 Mwt, and at 100 Mwt a pressure ramp would be made from 950 psia to 600 psia so that the turbine can be operated.

The program calls for TE-301 temperatures to be maintained by variation of feedwater flow. TE-301 is alarmed at 535 F and if it exceeds 550 F, the pump in the affected loop is tripped, with a back-up trip point at 575 F. If only two loops are operative and one of the pumps is tripped, this results in a reactor scram.

2. Observed Instabilities

On July 5 the reactor plant was at 67 Mwt on two loops, with 950 psia steam pressure, and superheated steam was being generated. A ramp to 600 psia was attempted but this resulted in peak-to-peak oscillations as high as 100 F on the TE-301 readings on one unit, causing its pump to shut down, which in turn scrambled the reactor.

On July 8 the reactor plant had been brought to 100 Mwt on three loops and was at a stable operating point in the superheat region with 950 psia pressure. However, manual control of feedwater flow had caused the sodium temperatures to overshoot while going up in power. When the TE-301 temperature in Loop 2 reached 540 F, it started to oscillate and this caused it to shut down. Shortly thereafter the same circumstances shut down Loop 1, resulting in a reactor scram.

3. Explanation

Under nominal conditions boiling should not occur in the downcomer portion of the steam generator tubes. However, it is possible that water header hydraulic imbalances or sodium flow maldistribution could cause the boiling to move up into the downcomer region on certain tubes. This would cause flow starvation in these tubes, and could result in most of the flow going through only a few tubes, with almost complete loss of heat removal capacity being the final consequence.

A series of computer programs were run using the conditions of Paragraph 1 above except steam pressure was varied from 600-950 psig and TE-301 temperature from 475-517 F. The most significant values obtained were the preheater areas necessary to bring the water in the tubes to saturation temperature. These values are plotted on page 34. The asterisks indicate the July 5 and 8 operating points; the preheater area required to initiate boiling was 1500 sq ft in each case.

The total heat transfer area in the downcomer section is estimated to be 1700 sq ft. (The computer program assumes no heat is transferred in the downcomer section.) Furthermore, a reduction of flow of only ten percent at the assumed points of instability would reduce the required preheater area by 20 percent, thus moving the point of the onset of boiling further into the downcomer region.

It was concluded that stable operation should be possible as long as the preheater area is held in excess of 1700 sq ft. In order to allow for pressure fluctuations and inaccuracies in TE-301 control, it was tentatively decided to retain a nominal operating point of 2000 sq ft or more.

4. Thermal Inertia

During all runs involving operation under conditions where superheated steam is generated, it was observed that the rate at which superheat is obtained or lost is rather fast. The rate seems to have no relation to the rate of rise or decrease in power.

One explanation is that thermal inertia type phenomena may be the cause. Temperature changes as seen by the instruments in the steam bypass line may not be related to physical conditions in the tubes at the same time. Special thermocouples have been installed in some of the tubes and in the headers of the No. 1 steam generator and preliminary data was being obtained during the October 5 operation. Additional experimental information will be obtained when reactor operation is resumed.

ESTABLISHMENT OF STABILITY CRITERION BASED ON PREHEATER AREA

SECTION V A

5. Future Operating Conditions

- a. Pressure ramps will be eliminated.
- b. 2000 sq ft of preheater area, minimum (as described in Paragraph 3 above).
- c. Sodium flows and temperatures as well as steam pressures and temperatures will be balanced in deciding the correct conditions for the circumstances. Some of the items to consider are: plant efficiency, core outlet temperatures, maximum desirable sodium flows, minimum desirable sodium temperatures, number of loops in service, reactor vessel temperature differentials, steam dump line capacity, intermediate heat exchanger effectiveness, and flexibility of turbine operation.

The following list represents one set of operating conditions which has been proposed for 200 Mwt operation:

Estimated Blanket Flow Rate	13%
Flows (10^6 lb/hr/loop)	
Primary Sodium	2.95
Secondary Sodium	3.75
Feedwater	0.22
Temperatures (F)	
Reactor Inlet	550
Bulk Reactor Outlet	800
Steam Generator Inlet	695
Steam Generator Outlet	500
Steam Conditions	
Pressure (psia)	800
Superheat (F)	157
Number of Loops in Service	3

During the three-day run in August 1966, at a constant 100 Mwt the stability of the turbine operation with the following steam conditions was successfully demonstrated:

Estimated Blanket Flow Rate	13%
Flows (10^6 lb/hr/loop)	
Primary Sodium	2.75
Secondary Sodium	2.60
Feedwater	0.11
Temperatures (F)	
Bulk Reactor Outlet	670
Reactor Inlet	545
Steam Generator Outlet	500
Steam Conditions	
Pressure (psia)	850
Superheat (F)	110

SECTION VI

MAINTENANCE AND ADDITIONS

A. Thawing the Reactor Vessel Syphon Drain Line

Provision was made in the plant design to permit draining of the reactor vessel if it should ever become necessary. The drain line connects to the storage tanks and extends through the 7 ft thick reactor operating floor. For syphoning of the sodium from the vessel, a temporary line would be inserted into the vessel through the shield plug or exit port and tied into the permanent portion of the line at a flanged connection at the operating floor. See the elevation view of this line on page 37.

As a result of the October 5 fuel damage incident, it was considered desirable to prepare to lower the sodium level in the reactor if this should become necessary in order to evaluate the condition of the reactor subassemblies by visual inspection. However, the portion of the drain line in the reactor floor (unheated) had become plugged with frozen sodium several years ago because it had not been blown dry after use. The problem was to thaw the plug and blow the line free.

As shown on page 37, the 2-inch line is about 12 ft long and has four 90 degree ells where it passes through the floor. It has 1 inch of Min-K insulation.

The blind flange and a portion of frozen sodium were removed from the top of the line. A 2 kw, 480 v immersion heater was inserted in a hole in the blind flange which was then reinstalled. A thermocouple was installed on the outside of the pipe. It was connected to an automatic controller which cycled the heater so as to maintain 650 F.

In order to fill the void space around the immersion heater elements a pot of sodium was connected to the top of the 1/2-inch vent line which branches off from the top of the 2-inch drain line. The pot was heated with acetylene gas and the 1/2-inch line was electrically heated. Argon at 10 psig pressure was maintained over the sodium in the pot, first to force the sodium down around the immersion heater and then to blow the 2-inch line free after it thawed.

The remainder of the syphon drain line, outside the floor penetration, was heated. The necessary valves were opened so that there would be a clear flow path for the sodium after it all became liquid. After about 12 hours the line blew free, as indicated by a drop in the sodium pot cover gas pressure.

REACTOR SYPHON LINE FLOOR PENETRATION