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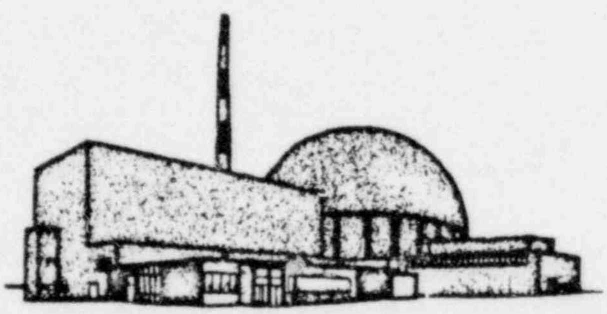
ANNUAL REPORT
YEAR 1962



DRESDEN NUCLEAR POWER STATION

Commonwealth Edison Company

R

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION

ANNUAL REPORT OF STATION OPERATION FOR THE YEAR 1962

January 18, 1963

DRESDEN NUCLEAR POWER STATION

ANNUAL REPORT OF STATION OPERATION

References:

(1) "Increase in Primary and Secondary Steam Flow," forwarded to AEC under cover letter I. L. Wade to Mr. Robert Lowenstein dated September 7, 1961.

(2) "Authorization of Increase in Primary and Secondary Steam Flow Withheld Pending Additional Information," AEC letter Mr. Robert Lowenstein to Mr. Murray Joslin dated October 2, 1961.

(3) "Additional Information on Proposed Increase in Primary and Secondary Steam Flow Rates" forwarded to the AEC under cover letter I. L. Wade to Robert Lowenstein dated January 5, 1962.

(4) Application for amendments to License DPR-2 Appendix "A" forwarded to the AEC under cover letter I. L. Wade to Robert Lowenstein dated January 5, 1962.

(5) Letter dated January 11, 1962, Walter G. Belter to I. L. Wade, requesting operating data on radioactive waste handling at Dresden.

(6) Letter dated January 24, 1962, I. L. Wade to Walter G. Belter, transmitting liquid, dry and spent resin waste data, and paper by Warren Kiedaisch entitled "Experience with Liquid Waste Handling at Dresden Nuclear Power Station."

(7) Letter dated January 31, 1962, I. L. Wade to F. P. Baranowski outlining additional costs that would be incurred by Commonwealth Edison Company in order to provide on-site storage of spent fuel.

(8) Request dated February 9, 1962, for approval of procedures regarding employe exposure forwarded to AEC under cover letter I. L. Wade to Robert Lowenstein.

(9) Request dated February 26, 1962, I. L. Wade to Robert Lowenstein, for consideration of three amendments to Byproduct Material License 12-5650-1 (H-63); (a) Clarification of Subitem (e) of Condition 20 Requiring Modification of Item No. 6 of "Supervisory Procedure Governing Use;" (b) Approval for Use of Additional Respiratory Protective Equipment; (c) Change in the Time Limit for Wearing of Face Masks.

(10) Letter dated March 2, 1962, I. L. Wade to Robert Lowenstein, transmitting additional information requested by R. S. Boyd by telephone on February 14, 1962.

(11) Letter of request dated March 2, 1962, I. L. Wade to Robert Lowenstein, pursuant to paragraph 3.a.(4) of License DPR-2, requesting amendment of Item No. 3, "Determination of Maximum Reactor Power" of Section D, Power Operation; including Supporting Information and Hazard Evaluation Respecting the Application for Increase in Heat Flux Limits.

(12) Letter dated March 15, 1962, R. L. Kirk to Murray Joslin requesting further additional information.

(13) Letter dated March 22, 1962, I. L. Wade to Robert Lowenstein, requesting amendment of Appendix "B" of License DPR-2, as amended, paragraph 4; also revision of Appendix "B" to conform with the attached chart, "Summary of 40-Year Special Nuclear Material Requirements" dated December 31, 1961.

(14) "Applicability of Burnout Correlation to Dresden Fuel," attachment "A" of letter dated March 27, 1962, forwarded to the AEC under cover letter I. L. Wade to Robert Lowenstein.

(15) Letter requesting amendments to application of January 5, 1962 for eleven amendments to License DPR-2, as amended, including Attachment "A", "Revisions to Type II Fuel Reload Amendment", forwarded to the AEC under cover letter I. L. Wade to Robert Lowenstein dated March 27, 1962.

(16) Letter of March 23, 1962, Robert Lowenstein to Murray Joslin regarding letters of (I) July 11, 1961, I. L. Wade to Robert Lowenstein, concerning exemption from equipping high radiation areas with control devices for visible and audible alarms; (II) letter of August 9, 1962, I. L. Wade to R. H. Bryan, concerning additional information requested regarding letter (I); (III) letter of October 4, 1961, Robert Lowenstein to Murray Joslin, concerning additional information requested regarding letters (I) and (II).

(17) Letter of April 4, 1962, Robert Lowenstein to Murray Joslin regarding reactor leak rate test.

(18) Letter of April 10, 1962, I. L. Wade to Robert Lowenstein requesting increase in maximum reactor power to levels up to and including, but not in excess of 700 MW (thermal) and including a description and Hazards Evaluation Report dated April 6, 1962.

(19) Letter from AEC requesting more information on pending amendment to License SNM-225, dated April 12, 1962, Conald A. Nussbaumer to H. K. Hoyt.

(20) Letter dated April 17, 1962, I. L. Wade to Robert Lowenstein, submitting a revised application concerning exemption from equipping high radiation areas with control devices for visible and audible alarms as previously submitted in letters of July 11, 1961, I. L. Wade to Robert Lowenstein, except for Exhibit A and August 9, 1961, I. L. Wade to Robert Lowenstein.

(21) Letter regarding requirement that applications for amendment to license be served on local officials, Howard Shapar to I. L. Wade dated April 24, 1962.

(22) Letter dated April 25, 1962, I. L. Wade to Robert Lowenstein requesting that "Refueling Instrumentation" be added to Amendment No. 10, of reference dated January 5, 1962 requesting amendment of Appendix A of License DPR-2.

(23) Letter requesting additional information regarding increase in heat flux limits and power level, R. L. Kirk to Murray Joslin dated May 9, 1962.

(24) "Fifth Quarterly Report of Station Operation" dated May 15, 1962, forwarded to the AEC under cover letter I. L. Wade to Robert Lowenstein dated May 24, 1962.

(25) "Request for Authorization to Use the Shipping Cask S-1 Supplied by the Stanray Corporation" dated May 24, 1962, I. L. Wade to Robert Lowenstein.

(26) Letter dated May 28, 1962, I. L. Wade to Robert Lowenstein, transmitting additional information on heat flux increase and power level increase.

(27) "Modification of Hydraulic Control Blade System - Hazards Analysis by the Research and Power Reactor Safety Branch" dated May 29, 1962, by Robert H. Bryan forwarded to Commonwealth Edison Company under cover letter by Robert Lowenstein to Mr. Murray Joslin dated May 29, 1962, granting authorization to incorporate this modification into the hydraulic control blade drive system.

(28) Correction letter to Reference No. 25, dated May 31, 1962, I. L. Wade to Mr. Robert Lowenstein.

(29) Letter dated June 1, 1962, R. Lowenstein from I. L. Wade, regarding proposed leak testing.

(30) Information letter to AEC entitled Refueling Instrumentation, I. L. Wade to Roger S. Boyd dated June 8, 1962.

(31) Letter of information concerning difficulty with one of the eighty control rod drives (C-7), H. K. Hoyt to Robert Lowenstein dated June 23, 1962.

(32) Letter from AEC authorizing increase in primary and secondary steam flow rates, R. Lowenstein to M. Joslin dated July 5, 1962.

(33) Letter regarding Shipping Cask, dated July 6, 1962, I. L. Wade to F. P. Baranowski.

(34) Notice of Hearing on Amendment to License DPR-2, dated July 6, 1962, Woodford B. McCool to Commonwealth Edison Co.

(35) Letter from AEC transmitting ACRS Report, Robert H. Bryan to Murray Joslin dated July 13, 1962.

(36) Letter to AEC regarding increase in heat flux limits and power level, I. L. Wade to Robert Lowenstein dated July 13, 1962.

(37) Letter to AEC concerning rewrite of "Refueling Instrumentation" and Exhibit A, Reactivity Comparison of Dresden Poison Blades, I. L. Wade to Robert Lowenstein, July 16, 1962.

(38) Letter to AEC transmitting technical information on revision to moderator temperature coefficient, I. L. Wade to Robert Lowenstein, July 20, 1962.

(39) Document from AEC, Docket No. 50-10, dated July 20, 1962, Robert H. Bryan to Murray Joslin, including Hazards Analysis of Request for Increase in Maximum Heat Flux Limitations.

(40) Letter to AEC regarding respiratory protective equipment, dated July 30, 1962, I. L. Wade to Robert Lowenstein, including Exhibit "A".

(41) Letter from AEC, dated August 6, 1962, authorizing use of 108 Type II fuel assemblies, hazards analysis, Appendix "A" of License DPR-2 revised in its entirety in accordance with Commonwealth Edison Co. application of January 5, 1962, and Appendix "B" of License DPR-2 revised in accordance with Commonwealth Edison Co. application of February 20, 1962.

(42) Telegram authorizing revision of moderator temperature coefficient limits, R. Lowenstein to I. L. Wade, August 17, 1962.

(43) Letter of acceptance of request for exemption from paragraph 20.203 (c)-2 of 10 CFR 20 (concerns equipping high radiation areas with visible or audible alarms), Robert Lowenstein to Murray Joslin dated August 23, 1962.

(44) Letter of August 24, 1962, I. L. Wade to Robert Lowenstein, regarding Dresden core grid support structure, including Exhibits "A", "B" and "C".

(45) Letter to Murray Joslin from Robert Lowenstein, dated September 5, 1962, authorizing operation at 700 MWt (Item 3.a.(1) of License DPR-2).

(46) Letter R. Lowenstein to Murray Joslin, dated September 19, 1962, transmitting a complete revision of the Technical Specifications (Appendix "A") of License DPR-2.

(47) Letter I. L. Wade to Robert Lowenstein, dated November 2, 1962, containing additional information requested and replacements for requests of February 26, 1962 and July 30, 1962 for respiratory protective equipment.

(48) Letter to Robert Lowenstein from Murray Joslin, Cocked Rod Criterion in D.5.b. of DPR-2 dated November 15, 1962.

(49) Letter of December 11, 1962, R. Lowenstein to Murray Joslin transmitting Annex "A", "Additional Conditions Relating to Authorization to Commonwealth Edison Co. to make Allowance for use of Respiratory Protection Equipment".

(50) License No. SNM-638, dated December 28, 1962, authorizing use of the Stanray Cask.

(51) Letter I. L. Wade to Robert Lowenstein, dated December 31, 1962, Application for Use of Knapp Mills Shipping Cask, including Exhibit "A" and "B".

I. INTRODUCTION

This report is submitted in compliance with paragraph 3.c.(2) of License DPR-2, as amended, and covers operation of the licensed facility at Dresden Nuclear Power Station during the entire year of 1962. This is the first annual report submitted, being preceded by quarterly reports Nos. 1 through 5. Operation continued except during short shutdowns for minor maintenance and tests until the station was shut down for the first refueling and to conduct an inspection of the control rod drives, control rod blades, and the core support grid.

II. SUMMARY OF OPERATION

A. Scope of Operations

The station operated within the limitations of License DPR-2 for power generation purposes. Variations from normal power generation schedules were required for reactor operator training and AEC operators license examinations; to conduct reactor physics measurements in connection with startup rods-out critical predictions, temperature coefficient reactivity measurements, control rod following verification and reactivity worth tests, shutdown margin tests, reactor power distribution and core reactivity loss checks, Xenon transient tests; and for fuel leak detection and location tests, instrument calibration, radiation surveys and radiochemical analysis.

B. Shutdowns

The station was shut down a total of 10 times during the year. Two forced shutdowns occurred due to a scram at power. Seven planned shutdowns occurred for maintenance and testing, and the first partial refueling outage began on November 7. These shutdowns are all listed in Table 1.

C. Load Restrictions

Load restrictions were imposed a number of times as listed in Table 2. These were usually planned for testing or maintenance. Derating due to fuel depletion began on September 17 and continued to increase until scheduled shutdown for refueling. Control rod patterns were adjusted to reduce fuel element leakage subsequent to October 23, 1962, resulting in an additional load restriction until shutdown.

III. DISCUSSION

A. Operating Experiences

1. The total reactor operating (critical) time during the year was 7,043.7 hours and the total integrated power for the period was 168,007.9 MWD (thermal). Gross electrical generation was 1,249,600,000 KWH during the year

III. DISCUSSION (Cont'd)

A. Operating Experience (Cont'd)

1. (Cont'd)

and 2,080,376,200 KWH to date. Total reactor operating time above 100 psig was 7,603.9 hours subsequent to the October 1961 inspection outage (December 5, 1961 to November 11, 1962). During the year the reactor was brought to critical 167 times, 122 of these criticals occurred during the December, 1962 tests for Type II fuel uniformity and control rod blade tests and Type I fuel reactivity and depletion tests.

2. Scrams

Two scrams occurred while operating close to full power.

a. At 11:40 A.M. on August 16, 1962, during normal steady state power operation, a spurious scram occurred upon activation of the high neutron flux trips on channels No. 3, 5 and 6. At the time, there were 70 fully withdrawn and 8 partially withdrawn control rods. Reactor thermal power was 628 MW and electrical load was 194 MW.

The scram tripped the secondary load limit which closed the secondary steam control valves. Primary steam flow increased momentarily due to the coupling of the primary and secondary valves through the control mechanism.

The turbine trip relay was actuated by high primary drum level which tripped the primary control and stop valves, the secondary stop valves, and opened the main 138 KV breaker on the unit.

The instantaneous overload relay operated on transformer 11 which tripped the generator auxiliary relay and this in turn opened the main field breaker. The loss of voltage on the 4 KV bus 11 tripped out "A" and "B" reactor recirculating pumps. An operator switched the automatic transfer on transformer 12 to the "off" position and attempted to close transformer 12 to bus 11. Bus 11 was still alive at reduced voltage, and transformer 11 was still closed to bus 11. Transformer 12 to bus 11 tripped on time delay overcurrent and reduced the bus 12 voltage enough to trip "C" and "D" reactor recirculating pumps. Following the trip-out of transformer 11, bus 11 was closed to transformer 12. One condensate, one primary feed and one secondary feedpump remained in service.

III. DISCUSSION (Cont'd)

A. Operating Experience (Cont'd)

2. Scrams (Cont'd)

a. (Cont'd)

The probable cause of the reactor scram was a combination of events on the out-of-core micromicroammeter channels 3, 5 and 6. The recording charts on the micromicroammeters did not indicate an increase in flux, which means that the trips were probably caused by electrical transients of short duration or by a low supply voltage.

The motor generator set supplying "B" safety system was found to be unstable because of worn brushes and high mica on the exciter commutator. It appears that the voltage on "B" safety system went low at the same time a spurious trip occurred on channel No. 6. Trips on channels 3, 5 and 6 would cause a reactor scram because No. 3 and No. 5 are on "B" safety system and No. 6 is on "A" safety system.

b. At 9:37 P.M. on August 28, 1962, during normal steady state power operation, the reactor scrammed upon activation of low primary drum water level trips, which resulted from a chain of events as described below. At the time of the initial disturbance, reactor thermal power was 626 MW and electrical load was 191 MW. There were 70 control rods fully withdrawn and 8 control rods partially withdrawn.

At 9:35 P.M. out-of-core power range channel No. 1 tripped. A few seconds after resetting channel No. 1 trip, out-of-core channel 1, 3 and 5 tripped. The trips were a result of low supply voltage from the "B" system motor generator set. The "B" system trips would have only alarmed; however, a normally open valve on the control air to accumulator No. 8 was closed. When the "B" system trip occurred, the three fully withdrawn control rods on accumulator No. 8 inserted.

Following the three control rod scram insertion, the thermal power dropped from 626 MWt to about 490 MWt and then recovered to about 510 MWt. The turbo-generator load dropped from 191 MWe to 150 MWe and recovered to 155 MWe. At about the same time a trip signal was produced by in-core amplifier 110A and the operator inserted a control rod next to in-core 110A.

III. DISCUSSION (Cont'd)

A. Operating Experience (Cont'd)

2. Scrams (Cont'd)

b. (Cont'd)

The primary drum level dropped due to the three rod scram and the low drum level alarm annunciated at minus 7 inches. The level then began to rise very rapidly and the primary feedwater control was switched from auto to manual control. The switch was made when the operator decided the control was not operating properly. Then the drum level reached plus 4 inches, the turbine lockout relay tripped from high primary drum level. The primary and secondary control and stop valves were tripped and the 138 KV breaker tripped. The elasped time between the three rod scram and turbine tripout was 1-1/2 minutes.

The unit slowed down rapidly because of auxiliary power transformer 11 loading the generator with the voltage regulator trying to maintain normal voltage.

Following the turbine trip, the reactor pressure peaked at 1010 psig which caused the reactor power to peak at about 630 MWt. At 9:37 P.M. the reactor scrammed from low primary drum level (minus 16 inches) and the four reactor recirculating pumps were tripped from low drum level (minus 25 inches). (Values given from instrument zero which is 4 inches above drum center line.)

When the turbo-generator tripped, 4 KV bus 12, together with all equipment supplied from it, was lost. Prime equipment on 4 KV bus 12 was "C" and "B" condensate pumps, "B" and "C" primary feed pumps and "B" safety system M-G set.

Transformer 12 did not close automatically to 4KV bus 12. Transformer 11 to 4 KV bus 12 was opened manually and transformer 12 was closed to 4 KV bus 12 manually at 9:43 P.M.

Following the scram, the commutator on the "B" safety system motor-generator set exciter was dressed and new brushes were installed. All connections on the voltage regulator were checked and the field rheostat was cleaned. A thorough check of the feedwater control system was also made.

III. DISCUSSION (Cont'd)

A. Operating Experience (Cont'd)

2. Scrams (Cont'd)

c. Thirteen spurious scrams occurred between the period December 20 and December 31, 1962 inclusive, during the reactor core critical test operations. During this period, three in-vessel neutron sensitive chambers were located in the vicinity of the fuel changes, replacing three external chambers in the safety system and arranged so that any one would actuate a scram. Two of these provided for scram on high neutron level and one provided for scram on short period. At least one operating low level neutron monitor was provided in the vicinity of the core and not connected into the safety system during critical testing. This instrumentation is in accordance with the requirements of paragraph E.5., License DPR-2 (Ref. 46). The condition for which these monitors were set to actuate a scram was not reached or approached in any of these scrams. Cause of the trips and resultant scram resulted from jarring of the chambers during the work or from spurious electronic disturbances.

3. System Stability

There has been no evidence of any change in system stability. The system remained stable during all periods of operation, including "short core" tests conducted on March 9 - 10, 1962 (described in section III-A-11b. below), and during stability tests, conducted on November 6, 1962.

The stability tests were conducted at 160 MWe and included measurement by means of a high speed recorder, the changes in primary steam drum pressure, out-of-core neutron flux and in-core neutron flux. Separate tests involved cycling by intermittent insertion and withdrawal of one control rod (H-7), one notch by normal control rod movement; operation of electromatic relief valves; bypass valve operation, change of pressure set point on the controlling backup pressure regulator; and turbine trip from 78 MWe load.

During a severe storm on April 30, 1962, two reactor recirculating pumps tripped due to a 138 KV bus fault and loss of auxiliary transformer No. 12. Stability was demonstrated by good reactor and auxiliary systems response during an attendant loss of electrical load from the operating load of 194 MWe to a load of 161 Mwe which was further reduced to 140 MWe by operator action. Normal recovery of load was accomplished following restoration of recirculating pump operation. The various auxiliary control systems maintained conditions within normal range of values and the reactor was controlled within scram condition set points.

III. DISCUSSION (Cont'd)

A. Operating Experience (Cont'd)

3. System Stability (Cont'd)

During August and September, the force restored regulator operation was abnormal, causing some fluctuations in electrical load by its effect on the turbine control valves. Although these effects were well within the range of safe operation, some testing and corrective maintenance was performed and the condition was alleviated. Further work on the pressure control system is being done during the refueling outage in progress at the end of 1962.

4. Incidents

There were no incidents during the year which compromised the safety of continued operation.

5. Control Rod Drives

During all periods of operation all control rod drives were verified for blade following once each week (except drive C-7 which remained inserted subsequent to June 22 for reasons described below).

During each calendar quarter all control rod drives were scram tested and friction tested. Quarterly tests were conducted January 26, June 21 - 22, and on August 17. Additional scram testing was conducted on February 27 - 28, during one shutdown which exceeded 24 hours. The drives were friction tested and scram tested following shutdown for refueling and inspections on November 8 - 9.

Removal and inspection of seven drives scheduled for inspection was well under way at year end. (Refer to inspection summary in section III-A-12.)

On May 3, control rods A-4, D-2 and D-5 on accumulator No. 18 scrammed. All three drives were fully withdrawn at the time of the scram. The turbo-generator output dropped from 193 to 169 MWe. Following the scram, the scram inlet and outlet valves continued to oscillate between the open and closed position.

The cause of the trip and the continuous closing and opening of the scram valves on accumulator No. 18 was found to be a bad ground connection at the terminal block in the scram fuse panel. The connection ties the common scram solenoid return leads to a common ground.

III. DISCUSSION (Cont'd)

A. Operating Experience (Cont'd)

5. Control Rod Drives (Cont'd)

Having the ground open caused the two scram solenoids to be in series with both "A" and "B" safety system M-G sets. The M-G sets are non-synchronous and produce a beat voltage which varies between 0 to 230 volts at a frequency which varies with the load on the safety system. When the beat voltage was 230 volts, the two scram solenoids were energized.

The trouble was corrected by securing the ground spider to the terminal block.

Discussion of control rod drive C-7 malfunction (Ref. 31):

On June 21 during the shutdown of the reactor to perform quarterly drive tests, drive C-7 (Serial No. 1232), which had been in position 12, stuck in position 5 and could not be moved in or out. Following insertion of all remaining control rods, C-7 was finally inserted to position 0.

During friction testing, great difficulty was had in moving C-7 to position 12. In friction testing the drive, it moved from position 12 to 6 at which point it stopped. Attempts to move the drive failed; after repeated attempts using various pressure combinations, the drive was moved to position 4.

At this time, a test program was initiated to evaluate the problem.

The companion drives, B-3 and E-4, on accumulator No. 20 were friction tested with good results. Various valving arrangements were tried in an attempt to indicate any defective valves. The Asco valves on C-7 were disassembled and rechecked for operation and condition. All valves appeared in good order. Following reinstallation of these valves, it was possible to move the drive to position 0.

Attempts were made to exercise the drive but the drive stopped in position 3 and could not be moved. The Ascos were again disassembled and it was noted that the water draining from the valve was dirty.

The drive was suspected to have a dirt problem where either the dirt had caused galling of the index tube when passing through the guide bushing or dirt in the ball shuttle valve which bypassed the water for the drive into the reactor.

III. DISCUSSION (Cont'd)

A. Operating Experience (Cont'd)

5. Control Rod Drives (Cont'd)

On June 22 a flushing system was set up using Pr + 200 water. The system was such that flushing could be made either through the up or down port. After several hours of trying various combinations of flushing, the drive was moved from position 3 to 0.

All attempts to move the drive from position 0 failed. The Asco valves were disarmed and the drive was put out of service.

It was concluded that the cause of the inability to move the drive was due to excessive friction, possibly caused by galling of the index tube. Galling of the index tube had been observed in test operations of drives at San Jose, California.

It was decided that the drive would be left fully inserted in a deactivated condition and plant operations would be resumed.

During this same outage friction and scram testing of all drives, with the exception of C-7, was conducted. When tested control rod drive J-3 (Serial No. 1250) indicated a decrease in the time in the buffer region over that obtained in February 1962 by a factor of two. The drive design is such that effective buffer is required only at low vessel pressure when the accumulator is supplying the drive force for scram. At high reactor vessel pressures, the accumulator is not effective at the end of the stroke and driving force is supplied by differential of vessel pressure to vent pressure.

On this basis, General Electric Company recommended that scrams at zero reactor pressure on drive J-3 be held to a minimum, but no restriction be placed on operation of the drive above 500 psig.

6. Control Rod Blades

During all periods of operation all control rod drives have been verified for blade following once each week, with the exception of C-7, after June 22, 1962. During each startup control rod blade patterns for criticality have been predicted and all blades verified.

Monthly control rod worth tests were conducted each month during operation. Boron concentrations in the reactor water remained low during the entire year.

III. DISCUSSION (Cont'd)

A. Operating Experience (Cont'd)

6. Control Rod Blades (Cont'd)

Critical tests were conducted in a small core configuration on December 28 to check poison strength of two irradiated control blades and compared to a "standard" unirradiated blade. Poison strength of the irradiated blades remained essentially unchanged.

7. Liquid Poison System

The liquid poison system was operative at all times during the period. The boron poison system was recirculated and sampled five times in the first calendar quarter and on June 23, August 18, October 19, November 9 and December 16. There were no conditions which would indicate a loss of boron from the solution tank.

8. Radioactive Waste Disposal

Release of radioactive liquid waste was accomplished in batch quantities at controlled release flow rates according to established procedures. The contribution to the activity of dilution water was always maintained within the limits specified in the applicable State and Federal regulations. The average contribution to the unidentified activity in the water utilized for radioactive liquid waste dilution during the year is calculated to be 0.0805×10^{-7} uc/ml (8.04 uuc/liter) compared to an average limit of 0.10×10^{-7} uc/ml (10 uuc/liter) for unidentified mixtures specified in CFR Part 20.

Solid radioactive wastes were stored on site pursuant to license DPR-2. One shipment of dry radioactive waste was trucked off-site on June 1, 1962 (1000 cubic feet of volume containing 500 millicuries of activity). Responsibility for this material was transferred to Isotopes Sales Department, Oak Ridge National Laboratory, under Permit Number B.F.E. 140.

One shipment of dry radioactive waste totaling 950 $ft.^3$ of volume and 400 mc of activity was shipped off-site on July 2. This material was transferred to Isotopes Sales Department, Oak Ridge National Laboratory, under Permit No. B.F.E. 1409 on July 2, 1962. Also, on the same date, one shipment of dry radioactive waste totaling 430 $ft.^3$ of volume and 560 curies of activity was transferred under Permit No. B.F.E. 1390.

One shipment of dry radioactive waste totaling 980 $ft.^3$ of volume and 127 mc of activity was shipped off-site on July 20, 1962 and transferred to Isotopes Sales Department of Oak Ridge National Laboratory under Permit No. B.F.E. 1409.

III. DISCUSSION (Cont'd)

A. Operating Experience (Cont'd)

8. Radioactive Waste Disposal (Cont'd)

Concentrations of noble fission product gases in the stack discharge to the atmosphere were well within license limits of 700,000 microcuries per second. The maximum rate of noble fission gas activity release (including all gas of long-lived activity) during each calendar quarter was 2,850, 3,600, 5,200 and 24,276 microcuries per second respectively.

9. Personnel Radiation Exposure

Personnel exposures to radiation were within limits as specified in 10 CFR Part 20.

10. In-Core Monitors

During all periods of reactor power operation, the in-core monitoring system was in service and in the safety circuitry within the requirements of License DPR-2.

11. Tests

a. Vertical Shape Test

Prior to plant shutdown in February a vertical power distribution test was conducted with no partially inserted control rods. The secondary steam flow was reduced to zero and the partial rods were repositioned to either fully inserted (8 rods at 0) or fully withdrawn (11 rods at 12). The results of the test indicated that Dresden will not be able to operate at full rated power will all control rods withdrawn near the end-of-core life.

b. Short Core Tests

A short core test was performed during March to simulate end-of-life conditions. The test provided fundamental data regarding the effects of primary steam generation and void distribution on vertical power distribution and in general the response of the vertical power distribution and core power to both group and bulk rod insertions. The test was also performed to check the ability and applicability of the SPUD and BORE computer programs to predict reactor core behavior at end-of-life.

III. DISCUSSION (Cont'd)

A. Operating Experience (Cont'd)

11. Tests (Cont'd)

b. Short Core Tests (Cont'd)

The test consisted of a rejection of all secondary steam flow and corresponding load reduction to approximately 120 MWe followed by the elimination of all partial rods first by the insertion of all rods which were initially set at levels less than six (6), second by the withdrawal of all rods initially set at position six or greater, and finally by the progressive insertion of all fully withdrawn rods, notch by notch (bulk insertion), until all were set at position six or less.

The vertical power distribution, resulting from the rod pattern changes necessary to accomplish the test were determined utilizing in-core readings and wire irradiations.

The vertical power distribution peaking factors increased from about 1.5 to almost 2.0 on the average as the partial rods were eliminated. This indicated that bottom entry rods could reduce vertical peaking factors resulting from the void distribution by at least 33 percent since vertical peaking factors of 1.5 or less can usually be maintained by proper rod positioning.

c. Vertical Shape Test

A series of nuclear checks were performed at Dresden on May 11 and 12, 1962, to obtain control rod power calibration data in both normal and test patterns, to determine the number of fully withdrawn rods needed to maintain rated primary steam flow rates, and to measure axial power distribution in simulated end-of-life states with a minimum number of partial rods.

The tests indicated that severe flux peaking in the lower portion of the core results with no partially inserted rods. Therefore it was estimated that about 8 partially inserted rods would be required at end-of-life. On this basis, it was calculated that derating would start about September 15, 1962.

A new rod pattern was established on May 11, 1962 prior to the performance of the Vertical Shape Test. It provided for the withdrawals of control rods necessary to attain our estimated end-of-life pattern with no major changes other than that required to equalize assembly exposures by withdrawal pattern reflection.

III. DISCUSSION (Cont'd)

A. Operating Experience (Cont'd)

11. Tests (Cont'd)

d. Temperature Coefficient Checks

The reactor was brought to critical at 4:12 P.M. on June 25, 1962. The successive withdrawal of forty-five (45) rods and three (3) notches were required to attain critical with a reactor water temperature of 138 F. An initial heating rate of approximately 86 F per hour was established subsequent to the verification of all unseen rods. It was noted that both power level and heating rate did not level off as usually experienced during previous xenon free startups. The continued increase in power level required the successive insertion of notches to counteract this positive characteristic. This phenomena has been noted during xenon decay startups but never during xenon free startups.

Reactivity measurements were made over the ranges of 250 to 300 F, 320 to 350 F, and 420 to 440 F. Measurements at lower temperatures was not possible due to the rapid heating rate required. All positive increments in temperature resulted in losses in reactivity as expected. Extrapolation of the data to lower temperatures indicated that the temperature coefficient was negative and too small to measure at 212 F. This extrapolation also indicates that the coefficient is positive below 190 F, as was evidenced during the June 25 startup by the progressive insertion of eight notches in a period of 30 minutes.

Based upon the temperature coefficient data, and particularly that obtained during the June 25 start-up, and other operating data, it was anticipated that the temperature coefficient of reactivity would become positive over a large portion of the temperature range before the reactor was shut down for refueling. As a result of these new tests, a proposal (Ref. 38) was submitted to the Atomic Energy Commission requesting a change to permit a positive temperature coefficient below 550 F. This submittal was approved on August 17, 1962 (Ref. 42).

Positive period reactivity measurements were made on August 19, 1962, with a fixed rod pattern at three discreet reactor water temperatures; 115, 175 and 200 F. The temperature coefficient over this range was found to be slightly positive.

III. DISCUSSION (Cont'd)

A. Operating Experience (Cont'd)

11. Tests (Cont'd)

d. Temperature Coefficient Checks (Cont'd)

The initial critical rod pattern was essentially maintained over the range of 155 to 280 F. A new rod pattern with less notches withdrawn were required to establish a pattern by which the temperature coefficient could be determined in the range of 280 to 360 F. The temperature coefficient appeared to change sign from positive to negative at 310 F. The "doppler" or metal temperature coefficient probably caused this apparent turn-over point at 310 F. A more probable turn-over point is 340 F based on a linear extrapolation of the reactivity change between the 280 and 365 F period determinations.

The coefficients as determined by the reactivity measurements over the temperature ranges tested are summarized by the following tabulation:

Temperature Range	Temperature Coefficient, $\Delta\rho/^{\circ}F$
155°F to 175°F	$+ 0.18 \times 10^{-5}$
278°F to 368°F	$- 1.32 \times 10^{-5}$
442°F to 458°F	$- 6.18 \times 10^{-5}$

e. Increased Primary Steam Flow Tests

On September 7, 1961 a request was made to the Atomic Energy Commission to permit increases in the primary and secondary steam flow rates (Ref. 1). The reasons for requesting the increases were as follows:

i. Increase turbine efficiency with increased ratio of primary to secondary steam.

ii. Increase operating flexibility by allowing a fixed rod pattern as reactivity decreases with burnup. Electrical load would be maintained by increasing secondary steam flow.

iii. Extending the life of the core without the necessity of derating the plant.

III. DISCUSSION (Cont'd)

A. Operating Experience (Cont'd)

11. Tests (Cont'd)

e. Increased Primary Steam Flow Tests (Cont'd)

On July 5, 1962, the Atomic Energy Commission authorized Dresden to increase the primary steam flow from 1,410,000 to 1,600,000 lb/hr and the secondary steam flow from 1,190,000 to 1,450,000 lb/hr (Ref. 32). The reactor power increase to 700 MWt requested April 10, 1962 (Ref. 18) had not been authorized by the AEC at the time of the tests described below.

On July 14, 1962 tests were conducted utilizing the increase in primary steam flow. Station instruments were used throughout the test. The major objectives of the tests were to determine plant parameters when operating with various combinations of primary and secondary steam flows, and any equipment limitations which would prevent the attainment of high primary steam flow.

Four test points were obtained as follows:

i. Primary steam flow of 1,600,000 lb/hr with sufficient secondary steam flow to obtain a reactor power of 630 MWt.

ii. Primary steam flow of 1,600,000 lb/hr with zero secondary flow.

iii. Primary steam flow of 1,500,000 lb/hr with zero secondary flow.

iv. Primary steam flow of 1,400,000 lb/hr with zero secondary flow.

The first part of the test consisted of increasing the primary flow to 1,600,000 lb/hr and then by the use of secondary steam increase the reactor power to 630 MWt. The increase in primary steam flow above 1,410,000 lb/hr was carried out in increments of 60,000 lb/hr. No difficulty was encountered with any equipment in the primary system.

Secondary steam flow was then increased and with reactor power near 630 MWt, Number 1 steam bypass valve opened to the condenser. At this point the primary control valves were in their maximum open position. Primary and secondary steam flows were then adjusted to keep the bypass valve from opening.

III. DISCUSSION (Cont'd)

A. Operating Experience (Cont'd)

11. Tests (Cont'd)

e. Increased Primary Steam Flow Tests (Cont'd)

An analysis of the test data showed that increasing the secondary steam flow with wide open primary control valves increases the primary bowl pressure. The effect of increased primary bowl pressure is to increase the steam pressure to the control valves, i.e., line pressure, and this in turn will cause a steam bypass valve to open to the condenser. What has taken place is that the head end of the turbine reached its flow passing ability. This condition is being improved by increasing the flow area at the high pressure turbine to increase flow capability to take a vantage of the increase in reactor power limit approved by the AEC on September 5, 1962 (Ref. 45).

During the tests the motor operated isolation valves on the spill lines to the condenser were closed. Past tests have shown that the air operated spill valves leak. The effect of this leakage is eliminated by closing the spill line isolation valves.

With high primary and secondary steam flow, difficulty was experienced in establishing drainage flow from the feedwater heaters. In particular the drain valve from "D" to "C" heater could not handle the flow with the result that "D" heater flooded. Modification in the feedwater drain system to take care of this situation is being made during the refueling outage.

The tests conducted on July 14 verified that the plant could be operated with high primary steam flow.

f. Increased Secondary Steam Flow Tests

On August 1, 1962, a test was conducted utilizing the increase in secondary steam flow as authorized by the Atomic Energy Commission on July 5, 1962 (Ref. 32).

The objectives of the test conducted on August 1, was to determine the plant parameters when operating with high secondary steam flow.

Station instruments were used throughout the test. The reactor power was limited to 630 MWt.

III. DISCUSSION (Cont'd)

A. Operating Experience (Cont'd)

11. Tests (Cont'd)

f. Increased Secondary Steam Flow Tests (Cont'd)

Primary steam flow was first reduced to approximately 1,150,000 lb/hr and the secondary steam flow increased to 1,200,000 lb/hr. Further increase in secondary steam flow above 1,200,000 lb/hr was carried out in increments of 50,000 lb/hr.

A maximum secondary steam flow of 1,420,000 lb/hr was reached with full secondary control valve opening. Primary steam flow was increased until a reactor power of 630 MWt and 1,410,000 lb/hr secondary steam flow was obtained. Primary steam flow at this point was 1,220,000 lb/hr. The results of the test are summarized below.

Primary Flow, lb/hr	1,220,000
Secondary Flow, lb/hr	1,410,000
Measured Gross Kilowatts	187,524
Measured Net Kilowatts	179,037
Test Gross Heat Rate	11,483
*Corrected Net Heat Rate	11,819
*Corrected Gross Kilowatts	190,843

*Corrected to 1.5 inch Hg exhaust pressure.

The motor operated isolation valves on the spill lines to the condenser were closed. Past tests have shown that the air operated spill valves leak. The effect of this leakage is eliminated by closing the isolation valves. It was not possible to get the heaters to drain properly. In particular, the drain valve from "D" to "C" heater could not handle the flow. It was necessary to complete the test with the motor operated isolation valves open. No other limitations were encountered.

g. Increasing Reactor Power to 700 MWt

On March 2, 1962, a proposal was made to the Atomic Energy Commission (Ref. 11) to increase the maximum heat flux for the fuel assemblies in the present core. This was followed by a proposal on April 10, 1962 (Ref. 18), for authorization to operate the reactor up to and including 700 MWt.

III. DISCUSSION (Cont'd)

A. Operating Experience (Cont'd)

11. Tests (Cont'd)

g. Increasing Reactor Power to 700 MWt (Cont'd)

On September 5, 1962, authorization was received from the Atomic Energy Commission (Ref. 45) to operate the reactor at power levels not in excess of 700 MWt, and to increase the maximum heat flux limits of the fuel.

On September 5, 1962, tests were conducted in bringing the reactor up to the higher power. Station instruments were used throughout the tests.

The tests had two major objectives:

i. Determine the maximum load carrying capability of the turbo-generator. The test would demonstrate the physical limitations of the turbine or any of its supporting components.

ii. Operation of the reactor at a thermal power of 700 MWt.

The first phase of the test program consisted of obtaining the maximum turbo-generator output with a maximum ratio of primary to secondary steam flows.

The thermal power of the reactor was increased to 645 MWt by increasing primary steam flow to 1,540,000 lb/hr and secondary steam flow to 1,170,000 lb/hr. Secondary steam flow was then slowly increased until the position lights of the primary control valves indicated wide open. As secondary steam flow is increased, the primary bowl pressure will increase and this in turn will cause the primary valves to open. A check of the primary bowl pressure indicated that any additional increases in secondary steam flow would open a by-pass valve to the condenser.

The gross electrical output at this point was 202,500 KW with a thermal power of 661 MW. No difficulti were experienced with any plant equipment.

The second phase of the test program consisted of obtaining a reactor power of 700 MWt with high secondary steam flow.

III. DISCUSSION (Cont'd)

A. Operating Experience (Cont'd)

11. Tests (Cont'd)

g. Increasing Reactor Power to 700 MWt (Cont'd)

It was first necessary to reduce the primary steam flow from 1,540,000 to 1,520,000 lb/hr. Secondary steam flow was then slowly increased. With a secondary steam flow of 1,340,000 lb/hr, both primary and secondary control valves were wide open. The gross electrical output was 209,032 KW with a thermal power of 675 MW. An analysis of the test data indicated that the flow passing capability of turbine had been reached.

During the tests the motor operated isolation valves on the spill lines to the condenser were closed. Past tests have shown that the air operated spill valves leak. The effect of this leakage is eliminated by closing the isolation valves. Due to difficulties in handling the heater drains during the high primary steam flow test in July, a manual bypass valve was installed in parallel with the air operated valve from "D" heater to "C" heater. The use of this valve eliminated the problem and no difficulty was experienced with the feedwater system.

Aside from the turbine limitation, there were no equipment or system limitations during the tests.

The last phase of the test program consisted of increasing primary steam flow to 1,600,000 lb/hr and reactor power to 700 MWt by increasing secondary steam. For this test, it would be necessary to bypass approximately 60,000 lb/hr of steam to the condenser. The purpose of this test was to reveal any limitations in the feedwater system.

Primary steam flow was increased to 1,560,000 lb/hr with No. 1 bypass valve opened 75%. Attempts to increase the primary steam flow resulted in a slow swinging of the bypass valve between open and closed. Movement of the bypass valve in turn caused the primary drum level to vary, which in turn caused swings in the primary feedwater system flow.

The test was terminated because of approaching a low suction pressure to the primary feedwater pump during the peak high flow rates. A third (spare) condensate pump could have been operated to increase feedwater

III. DISCUSSION (Cont'd)

A. Operating Experience (Cont'd)

11. Tests (Cont'd)

g. Increasing Reactor Power to 700 MWt (Cont'd)

pump suction pressure, but only two were operated to simulate normal operating conditions. The results of the tests are summarized below.

Test Point	1	2
Primary Flow. lb/hr	1,550,000	1,520,000
Secondary Flow, lb/hr	1,200,000	1,340,000
Measured Gross Kilowatts	202,500	209,032
Measured Net Kilowatts	194,154	200,573
Reactor Thermal Power, MWt	653.8	675.8
Test Gross Plant Heat Rate	11,009	11,063
*Corrected Net Plant Heat Rate	11,292	11,395
*Corrected Gross Kilowatts	205,920	211,499

*Corrected to 1.5 inches Hg.

All out-of-core micromicroammeters and all in-core fission chambers had been reset prior to the test to indicate percent of 700 MWt and 320,000 Btu/hr-ft^2 for Type I fuel respectively during the test. A slight re-adjustment of the control rod pattern was required during the test to attain the primary steam flow conditions due to the core fuel reactivity loss prior to scheduled refueling.

h. Xenon Decay Tests

In order to improve estimates for rod withdrawal to critical at various temperatures and xenon poison conditions, xenon poison effects on the control rod withdrawal requirements are determined whenever shutdown or startup conditions permit attaining useful data.

The July 16, 1962 startup, approximately 20 hours after shutdown for turbine extraction maintenance, required the withdrawal of 54 rods and 4 notches to attain critical. Twenty-seven (27) of these rods were unseen during their subcritical withdrawal. All unseen rods were verified, however, subsequent to the attainment of the critical state.

A constant reactor water temperature was maintained during the initial stages of reactor operation to determine the rate of xenon decay. A gain in reactivity equivalent to one notch per hour was noted at 290°F. This was surprisingly small compared to

III. DISCUSSION (Cont'd)

A. Operating Experience (Cont'd)

11. Tests (Cont'd)

h. Xenon Decay Tests (Cont'd)

that experienced on July 27, 1961 under similar conditions during which a 6 notch insertion was required. This difference may be due to the rods in use to control reactivity, i.e., their relative worths as affected by position in the core or setting, or as affected by exposure.

The rate of xenon decay and rise in temperature during heating was such that no additional rod withdrawals were required to attain near rated temperature. This phenomenon was first experienced during our last xenon decay startup of March 18, 1962.

i. Fuel Leakage Tests

Fuel leak tests were conducted prior to shutdown utilizing an extension of the usual procedures for monthly control rod worth tests. This permitted a close check of controlled local neutron flux changes attendant with control rod movement upon the off-gas and stack gas activities. As a result of these tests, several areas in the core were listed as suspect areas for leaking fuel. These areas were checked following shutdown and head removal by sampling water from individual assemblies and analyzing for iodine concentrations. As a result of these tests, nine fuel assemblies were determined to have defects in the cladding.

j. Stability Tests

These tests, conducted November 6, 1962, were previously described in this report in section III.A.3.

k. Critical Tests with Reactor Head Off

After completion of removal of fuel sufficient to clear a large central portion of the core for critical testing, control rod drive tests completed and systems checkout was completed, in-vessel instrumentation was installed and checked out and the reactor core was loaded to minimum critical with 28 new Type II fuel assemblies. The configuration was adjusted to provide the best arrangement for the critical tests to follow. Uniformity of all Type II fuel was

III. DISCUSSION (Cont'd)

A. Operating Experience (Cont'd)

11. Tests (Cont'd)

k. Critical Tests with Reactor Heat Off (Cont'd)

checked by repeated substitution of eight assemblies in the critical core lattice until all 106 assemblies had been tested on December 23. The tests revealed the expected uniformity did exist and that reactivity of new Type I and Type II fuels were equivalent in terms of reactivity.

The minimum critical assembly of Type II fuels was then expanded to 29 assemblies in a configuration which permitted suitable critical testing of 33 individual Type I irradiated fuels having a wide range of exposure. These tests permitted reactivity measurements and comparisons which will assist in calculation of fuel depletion (burnup). These tests were completed December 27, 1962.

The critical assembly was again expanded to permit critical testing of three control rod blades to determine variations in poison worth of two irradiated blades compared to a previously tested unirradiated "standard" blade. These tests indicated no measurable change or difference in reactivity poison worth of the irradiated blades compared to the "standard" blade. The tests were completed on December 28 after which the critical test fuel and the remaining spent fuel was unloaded. This work was completed December 30. On December 31, nine irradiated fuel assemblies were reloaded into the core as the initial steps of the refueling operation got under way.

l. Reactor Enclosure Air Lock Leakage Tests

The reactor enclosure air locks (equipment lock, personnel lock, and emergency lock) were checked for leakage on November 29-30, 1962. These tests were made subsequent to replacement of the plug-type equalizing valves in the equipment lock and personnel lock with an improved type of ball valve. These tests were conducted at 2 psig pressure using a Freon-12 tracer and flame color indicator. Only three small leaks were noted, all of which were on the equipment lock. One of these, having a 3/4" connection which is used for test and normally closed with a pipe plug was repaired by tightening the plug. The other two leaks were insignificant, representing less than 0.1 percent of the total allowable sphere leakage rate, and were not repaired (Ref. 17, 29).

III. DISCUSSION (Cont'd)

A. Operating Experience (Cont'd)

11. Tests (Cont'd)

m. Preoperational Mechanical Tests of Stanray Spent Fuel Shipping Cask

Preoperational mechanical tests involving handling of the cask and demonstration of its compatibility with fuel and fuel handling equipment in use at the Dresden site were conducted during November 1962 with satisfactory results. Application for use authorization had been submitted May 24, 1962 and approved December 28, 1962 (Ref. 25, 28, 33 and 50).

12. Inspections of Control Rod Drives, Blades and Core Grid Structure

These inspections, required by License DPR-2 to be conducted between 4500 and 9000 hours of operation above 100 psig subsequent to the previous inspection were conducted during the current refueling outage. Actual operating hours above 100 psig during this period were 7603.9. Details of these inspections will be forwarded to the AEC in a separate report; however, a brief summary is given below.

a. Inspection of the core grid structure was completed on December 9, 1962 and was conducted in accordance with the proposed procedure outlined in the submittal to the AEC dated August 24, 1962 (Ref. 44). The structure was found to be in good condition. Only one new crack was discovered, this being in a beam-to-beam weld in an area not previously inspected, which is well within calculated probability on the basis of cracks noted in previously inspected welds. Previously inspected cracks did not evidence any progression. Details of the inspection will be forwarded to the AEC in a separate report.

b. Inspection of eight irradiated control rod blades was completed on December 2, 1962. All blades inspected were found to be in good condition.

c. All seven control rod drives had been inspected by December 31, 1962. All were found in good condition, except C-7 drive, which had exhibited abnormal operation on the June 21, 1962 shutdown, and which remained inserted and out of service during the remaining operating period. C-7 was found to have a scored index tube resulting from chrome surfacing which had stripped from the collet lock collecting in the close clearance between the collet lock and index tube.

III. DISCUSSION (Cont'd)

A. Operating Experience (Cont'd)

13. Unloading for Refueling Inspections and Maintenance

Fuel unloading from the core started on November 19 and was completed on December 30. The six PF type developmental fuel assemblies and three Type I fuel assemblies which exhibited higher than average iodine concentrations during the "sipping" tests (described in section III.A. above) were the first fuel unloaded from the core to permit subsequent tests and inspections in the fuel building. A thorough analysis of the cladding failures is currently being made by General Electric Company. Their report on Type I fuel follows.

Recent fuel examination indicates that the Type I zircaloy clad fuel is experiencing occasional failure due to bowing of individual corner rods towards the channel which causes increased clad temperature and increased corrosion to the point of clad perforation. This type of failure seems to occur over a period of time and at random locations in the core. One such failure was observed after about 3000 hours of power operation and three similar failures were observed recently after 12,000 hours.

Upon inspection of the defective fuel removed from Dresden and analysis of the estimated performance which this fuel experienced it appears that:

i. While the reactor was operating at appreciable power levels the fuel rods became heated (not necessarily overheated) and by some mechanism (thermal bowing, axial growth, release of locked-up stresses due to cold working, etc.) moved within the assemblies. In the case of the higher power corner rods the motion brought the segments into contact or near contact with the channel surface.

ii. Due to inadequate cooling of the bowed rods while close to or touching the channel surface, the heat transfer mechanism became retarded and reverted from nucleate boiling to localized film blanketing of the fuel rod. The local part of the fuel rod closest to the channel box would be inadequately cooled and therefore would experience film blanketing.

III. DISCUSSION (Cont'd)

A. Operating Experience (Cont'd)

13. Unloading for Refueling Inspections And Maintenance (Cont'd)

iii. Analysis has indicated that this film blanketing caused a modest increase in the cladding surface temperature from about 570°F to about 800°F depending on the surface heat flux. These elevated temperatures are accelerating the process of the oxidation of the zircaloy cladding to the extent that clad perforation eventually occurs.

The damage to the fuel consists of oxidation of a saddle shaped region of the cladding surrounding the perforation in the clad wall. The size of two of the perforations was approximately 1/8 x 1/4 inch and the third was about 0.2 inch x 1 inch. Severe oxidation extended for an inch to several inches along one side of these tubes around the perforation.

In addition to the aforementioned defects, a second type of failure was detected in two rods of one assembly. These failures were characterized by localized bulging and cracking of the clad and were apparently due to high internal pressure associated with local clad defects. The breaks were clean and there was no associated corrosion or scale around the ruptures.

The consequences of both type of failures on the plant have been mild, the principal effect being increased release of offgas. At the end of the last operating cycle the total offgas release level was less than 10,000 microcuries per second. A major fraction of this total is considered to have emanated from several experimental fuel assemblies which were more severely damaged.

It is concluded from these observations that these failures of Type I fuel have no significant effect relating to plant safeguards and the fuel is satisfactory for continued operation.

Results of the analyses of the PF assemblies are not yet available.

Six non-defective PF Type and two Type II fuel assemblies scheduled for detailed examination were unloaded next. Subsequently, fuel assemblies were unloaded to permit control rod drive and blade inspection (32 assemblies), orifice removals (62 assemblies) and grid inspection (136 assemblies), critical testing in the central core region, gamma scanning for power distribution evaluation, and other spent fuels for reprocessing, in the order listed. The minimum number of fuel assemblies remaining in the core just prior to the start of refueling was 126, as a result of the removal of a total of 338 fuel assemblies from the core.

III. DISCUSSION (Cont'd)

B. Amendment to License DPR-2

1. Steam Flow Rates and Burnout Heat Flux Correlations

An application dated September 7, 1961 (Ref. 1) was made for license amendment to permit steady state operation of the reactor at higher steam flow rates and to permit use of improved burnout heat flux correlations. This application was approved on July 5, 1962 after several conferences with the AEC, and formal correspondence with the AEC (Refs. 1, 2, 3, 4, 10, 12, 14, and 46). Tests were conducted at high steam flow rates on July 14 and August 1, 1962, as described in section III.A. above, and operation conformed to the new burnout heat flux correlations immediately following the issuance of the amendment.

2. Increase in Power Level and Heat Flux Limits

An application for license amendment was made on March 2, 1962 requesting an increase in maximum heat flux limits at the over power condition for the various types of fuel in the Dresden reactor. Another application dated April 10, 1962 was submitted requesting authorization for an increase in maximum steady state power level from 630 MWt to 700 MWt. Additional information respecting each of these applications was submitted in response to AEC requests (Refs. 11, 18, 23, 26, 35, 36, 39, 45 and 46). Authorization was granted on September 5, 1962 for operation at 700 MWt and for higher heat flux limits for the various types of fuel.

High power tests were performed on September 5, described in section III.A. above with instrumentation adjusted in accordance with the newly authorized limits.

3. Type II Fuel Reload Amendment

An application for license amendment was made January 5, 1962, including requests for licensing the loading of a total of 108 Type II fuel assemblies, and for refueling instrumentation and procedures. Supplemental information was submitted March 27, April 25, June 8 and July 16, 1962 (Refs. 4, 15, 22, 30). Authorization was made by the AEC August 6, 1962 (Ref. 41).

4. Refueling Instrumentation

A request made April 25, 1962 (Ref. 22) to add "Refueling Instrumentation" to amendment No. 10 of the January 5, 1962 (Ref. 4) request for amendments to Appendix "A" of License DPR-2 and was supplemented by information to the AEC on June 6, 1962 (Ref. 30) and July 16, 1962 (Ref. 37) and approved by the AEC on September 19, 1962 (Ref. 46).

III. DISCUSSION (Cont'd)

B. Amendment to License DPR-2 (Cont'd)

5. Moderator Temperature Coefficient

A request for change in the requirements for moderator temperature coefficients was made on July 20, 1962 (Ref. 38) and was authorized by the AEC on August 17, 1962 (Ref. 42).

6. Special Nuclear Material Requests

Request was made March 22, 1962 to amend Appendix "B" of License DPR-2 for the 40 year special nuclear material requests (Ref. 13).

C. Notice of Changes Pursuant to Paragraph 3.a.(5) of License DPR-2

1. Control Rod Drive Hydraulic System

Approval was granted May 29, 1962 (Ref. 27) to incorporate modifications in the control rod hydraulic system requested July 11, 1961.

D. Application Made Under Part 20, 10 CFR

1. Exemption from Requirement of High Radiation Area Alarms

Letters regarding the request for approval of an exemption from equipping high radiation areas with control devices and audible alarms dated July 11, 1961 were exchanged on October 4, 1961, March 23, 1962 and April 17, 1962 (Refs. 16 and 20). A revised application was submitted April 17, 1962 and approved August 23, 1962 (Ref. 43).

2. Approval of Respiratory Protective Equipment

A request was made on February 26, 1962 (Ref. 9) for approval of amendments to Byproduct Material License 12-5650-1 (H-63) and certain classification of items of the December 6, 1961 approval, and approval of additional respiratory protective equipment. Additional information was submitted July 30 and November 2, 1962 (Refs. 40 and 47). Tentative approval was received December 11, 1962 (Ref. 49).

E. Compliance with Conditions of License DPR-2 Requiring Additional Information

1. Reactor Enclosure Leakage Rate Tests

Additional information regarding the reactor enclosure leakage rate test was requested by the AEC April 4, 1962 (Ref. 17) which was submitted June 1, 1962 (Ref. 29).

III. DISCUSSION (Cont'd)

E. Compliance with Conditions of License DPR-2 Requiring Additional Information (Cont'd)

2. Core Support Grid Structure Inspection

A proposal regarding inspection requirements for the core support grid structure was submitted August 24, 1962 (Ref. 44).

F. Applications Made Under Part 72 (Proposed) 10 CFR

1. Spent Fuel Shipping Cask

A request was made for authorization for use of cask S-1 supplied by Stanray Corporation May 24, 1962 (Ref. 25). A correction letter of May 31, 1962 was submitted (Ref. 28). Additional information was requested by the AEC July 6, 1962 (Ref. 33). License No. SNM-638 was issued by the AEC December 28, 1962 authorizing use of the Stanray Cask (Ref. 50).

Application was made for authorization to use the spent fuel shipping casks manufactured by Knapp Mills Company on December 31, 1962 (Ref. 51).

G. Other Reports and Approvals

Reference 5,6 - Radioactive Waste Handling Practices

Reference 21 - Local AEC advisement of applications for amendment to License DPR-2

Reference 7 - Fuel storage and reprocessing

Reference 31 - C-7 control rod drive problem

Reference 8 - Employee exposure

H. Modifications to Plant and Equipment

1. Reactor Enclosure Maintenance Scaffold

Work has proceeded on the installation of a large steel rotating scaffold designed and constructed by Chicago Bridge & Iron Co., on the exterior of the upper half of the reactor enclosure. The scaffold installation is essentially complete at year's end. The reason for its installation is to facilitate planned repair and replacement of most of the exterior thermal insulation on the upper hemisphere necessitated by separation of the insulation from the metal, and to permit future inspections and repairs. The scaffold, which weighs 26,230 pounds, rotates about a ring centered near the top located just beyond the emergency condenser vent and supported on 12 inch diameter pads welded to the sphere with 1/4 inch

III. DISCUSSION (Cont'd)

E. Compliance with Conditions of License DPR-2 Requiring Additional Information (Cont'd)

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III. DISCUSSION (Cont'd)

E. Compliance with Conditions of License DPR-2 Requiring Additional Information (Cont'd)

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A proposal regarding inspection requirements for the core support grid structure was submitted August 24 1962 (Ref. 44).

F. Applications Made Under Part 72 (Proposed) 10 CFR

1. Spent Fuel Shipping Cask

A request was made for authorization for use of cask S-1 supplied by Stanray Corporation May 24, 1962 (Ref. 25). A correction letter of May 31, 1962 was submitted (Ref. 28). Additional information was requested by the AEC July 6, 1962 (Ref. 33). License No. SNM-638 was issued by the AEC December 28, 1962 authorizing use of the Stanray Cask (Ref. 50).

Application was made for authorization to use the spent fuel shipping casks manufactured by Knapp Mills Company on December 31, 1962 (Ref. 51).

G. Other Reports and Approvals

Reference 5,6 - Radioactive Waste Handling Practices

Reference 21 - Local AEC advisement of applications for amendment to License DPR-2

Reference 7 - Fuel storage and reprocessing

Reference 31 - C-7 control rod drive problem

Reference 8 - Employee exposure

H. Modifications to Plant and Equipment

1. Reactor Enclosure Maintenance Scaffold

Work has proceeded on the installation of a large steel rotating scaffold designed and constructed by Chicago Bridge & Iron Co., on the exterior of the upper half of the reactor enclosure. The scaffold installation is essentially complete at year's end. The reason for its installation is to facilitate planned repair and replacement of most of the exterior thermal insulation on the upper hemisphere necessitated by separation of the insulation from the metal, and to permit future inspections and repairs. The scaffold, which weighs 26,230 pounds, rotates about a ring centered near the top located just beyond the emergency condenser vent and supported on 12 inch diameter pads welded to the sphere with 1/4 inch

III. DISCUSSION (Cont'd)

H. Modifications to Plant and Equipment (Cont'd)

1. Reactor Enclosure Maintenance Scaffold (Cont'd)

fillet welds. Two lower rings are supported from pads welded to the sphere circumferentially arranged just above the midplane and at a higher elevation about midway between the bottom and top rings. The upper of these two rings is supported on 20 inch diameter pads to the sphere whereas the lower ring is supported on 12 inch diameter x 1 inch thick carbon steel pads. Continuous 1/4 inch fillet welds are employed in accordance with ASME Unfired Pressure Vessel Code, paragraph UCS-56(e2). All welds are magnafluxed after welding without stress relief. The procedures are the same as followed for the Consumers Power Company Big Rock Plant reactor enclosure. The lightning rod was replaced with a new higher rod located at a 17 feet 6 inches radius from the center providing protection from lightning equal to the original rod. The rod is supported on a 24 inch diameter pad welded to the sphere.

2. Reactor Fuel Handling Canal

Major revisions planned for the canal include installation of two sumps to be located south of the reactor primary shielding. The sumps will extend down into the pipeway compartment. New shielding walls are in the process of construction for biological shielding in the localized area in the proximity of the sumps. The shielding extends from the pipeway floor at elevation 517 feet up to the under side of the canal floor. Both sumps will be supported from the canal floor which will be strengthened by reinforcing rod welded to existing floor reinforcing rod at the points of sump penetration of the floor. The larger sump is sized to accommodate the fuel basket. The smaller sump will accommodate vertical storage of reactor instrument in-core chamber assemblies.

The purpose of the sump installation is to reduce time required in the fuel transfer operation by eliminating the intermediate requirement for fuel rack transfers in the reactor canal. Secondly, a protective storage area is provided for the incore assemblies which have suffered a high mortality rate due to excessive repeated handling in previous outages, as necessitated by the need for water shielding during canal filling and draining for maintenance and fuel loading and unloading operations.

III. DISCUSSION (Cont'd)

H. Modifications to Plant and Equipment (Cont'd)

2. Reactor Fuel Handling Canal (Cont'd)

After completion of sump concrete work, the entire canal will be lined with 1/4 inch and 3/16 inch thick stainless steel to provide a water tight enclosure to eliminate leaks in the concrete walls and to facilitate cleaning of the canal walls after changes in water levels.

3. Control Rod Drive Temperature Monitoring

An entirely new temperature monitoring system is being installed, consisting of a more permanent thermocouple installation in each of the position indicating probes, and installation of temperature recorders. This will permit a more continuous monitoring of drive temperatures.

4. Main Condenser Circulating Water Piping

Because of repeated leaks in the concrete piping of this system causing some undermining of the sand fill in the adjacent areas, repairs are being made consisting of complete lining using an adhesive neoprene coating throughout. Pipe joints are further reinforced by circumferential internal steel rings which assist in holding the lining in place at the pipe joints. The system was tested and one leak which occurred will be repaired.

5. Removal of the Circulating Water Monitor

All of the water discharged to the river from the plant systems is monitored to insure that no release exceeds license limitations. This monitoring involves (1) batch processing, analysis and release at controlled rates of all liquid wastes discharged from the radioactive waste treatment plant, (2) continuous monitoring of "service water" used for heat removal in the plant process systems heat exchangers, and (3) continuous sampling and daily analysis of the composite sample of all water discharged from the plant to the river, including liquid radioactive wastes, service water, and the main steam condenser cooling water.

A continuous gamma scintillation monitor originally installed in the main steam condenser cooling water discharge line was removed from service after evaluation of the effectiveness and need of the monitor. Several improvements had been made to increase the sensitivity of the monitor without success. These

III. <u>DISCUSSION</u> (Cont'd)

H. <u>Modifications to Plant and Equipment</u> (Cont'd)

5. <u>Removal of the Circulating Water Monitor</u> (Cont'd)

improvements included relocation of the monitor and installation of supplementary shielding to reduce background radiation and modifications in the sampling system to improve sample flow. The problem associated with this type of monitoring has been accentuated by the extreme sensitivity required for the low level of activity likely in the event of a leak in the system and the large volume of dilution water attendant with the condenser cooling water flow.

TABLE I

REACTOR AND TURBINE GENERATOR SERVICE

Reactor					Turbine-Generator					Operating Condition.
On	Off	Critical Hr.	Critical Min.	Criticals	On	Off	Operating Period Hr.	Operating Period Min.	Starts	
	6:58 AM 1/ 7/62	150	58			8:07 PM 1/ 6/62	140	7		Power Operation. Shutdown for Turbine-Governor Inspection.
11:39 PM 1/ 7/62	1:50 AM 1/26/62	434	11	1	8:20 AM 1/ 8/62	9:59 PM 1/25/62	421	39	1	Power Operation. Shutdown for Turbine-Governor Maintenance and Thrust Bearing Inspection.
2:45 AM 1/26/62	6:00 AM 1/26/62	3	15	1						Xenon Following.
10:23 AM 1/28/62	3:32 PM 1/28/62	1	32**	5						Training Consumers Power Company Personnel
4:02 PM 1/28/62	11:59 PM 2/22/62	607	57	1	1:29 AM 1/29/62	11:50 PM 2/22/62	598	21	1	Power Operation. Shutdown for steam leak on in-core pressure seal.
5:08 AM 2/23/62	6:49 AM 2/23/62	1	40	1						Training Consumers Power Company Personnel
1:03 AM 3/ 1/62	11:41 AM 3/ 1/62	10	38	1						Training Consumers Power Company Personnel. Sub-critical for leak check under reactor.

TABLE I

REACTOR AND TURBINE GENERATOR SERVICE (Cont'd)

Reactor					Turbine-Generator					Operating Conditions
On	Off	Critical Hr.	Critical Min.	Criticals	On	Off	Operating Period Hr.	Operating Period Min.	Starts	
12:02 PM 3/ 1/62	1:39 AM 3/17/62	373	37	1	3:40 PM 3/ 1/62	1:32 AM 3/17/62	369	52	1	Power Operation. Shutdown for steam leaks in drum compartment.
10:42 PM 3/17/62	6:33 AM 6/21/62	2288	51	1	4:20 AM 3/18/62	4:58 PM 6/20/62	2267	38	1	Power Operation. Shutdown for quarterly friction and scram testing.
4:12 PM 6/25/62	12:27 AM 6/26/62	7	4**	7						Temperature coefficient measurements.
12:39 AM 6/26/62	11:13 PM 7/15/62	478	34	1	5:38 AM 6/26/62	11:10 PM 7/15/62	473	32	1	Power Operation. Shutdown for "D" extraction steam leak. Edison operator training.
7:15 PM 7/16/62	11:40 AM 8/16/62	736	25	1	4:31 PM 7/17/62	11:40 AM 8/16/62	727	09	1	Power Operation. Reactor scram. Remained shutdown for open house.
2:19 PM 8/19/62	9:10 PM 8/19/62	5	24**	6						Temperature coefficient reactivity measurements.
9:23 PM 8/19/62	9:37 PM 8/28/62	216	14		12:22 AM 8/20/62	9:37 PM 8/28/62	213	15	1	Reactor scram.

TABLE I

REACTOR AND TURBINE GENERATOR SERVICE (Cont'd)

Reactor					Turbine-Generator				Starts	Operating Conditions
		Critical		Criti-			Operating Period			
On	Off	Hr.	Min.	cals	On	Off	Hr.	Min.		
6:18 AM 8/29/62	6:08 PM 10/18/62	1211	50	1	3:56 PM 8/29/62	5:52 PM 10/18/62	1201	56	1	Power Operation. Shutdown to repair steam leaks in steam drum compartment.
6:47 PM 10/19/62	6:57 AM 11/8/62	469	10	1	11:20 AM 10/20/62	12:23 PM 11/ 7/62	433	03	1	Power Operation. Shutdown for re-fueling and turbine overhaul.
5:45 AM 11/ 9/62	6:31 PM 11/11/62	23	4**	16						Xenon following and temperature coefficient tests. Edison operator training and AEC exams.
11:20 PM 12/20/62	2:23 AM 12/28/62	23	20	122						Critical testing. Fuel uniformity and control blade exposure tests.
Total - 1962		7043	44	167			6846	32	9	

**Intermittent Operation

TABLE II

LOAD RESTRICTIONS

Date	Reduction in Electrical Capability - MWe	Condition
January 5 - 6	13	Control rod test pattern
January 12 - 13	15	Control rod test pattern
January 13 - 14	32	Control rod worth test
January 19 - 20	20	Control rod test pattern
January 25	25	Control rod test pattern
January 30	26	Maintenance on accumulator No. 2
February 2 - 3	27	Control rod test pattern
February 3 - 5	32	Control rod worth test
February 9 - 10	33	Control rod test pattern
February 10 - 12	32	Control rod worth test
February 13	32	Maintenance on accumulator No. 25
February 16 - 17	37	Control rod test pattern
February 17 - 18	32	Control rod worth test
February 18	52	Ice melting on transmission lines
March 1 - 7	34	In-core stabilization and calibration
March 7	129	Feedwater heater maintenance
March 9 - 10	169	"Short Core" test
March 10 - 11	34	Control rod worth tests
March 11 - 12	42	Control rod test pattern
March 16	53	Control rod test pattern
March 30 - 31		Control rod worth test
April 1 - 4	34	Control rod worth tests
April 5	34	In-core calibration
April 7 - 8	34	Control rod worth test
April 13 - 14	5	Control rod test pattern
April 20 - 21	11	Control rod test pattern
April 27 - 28	17	Control rod test pattern
April 30	52	Loss of Aux. Transformer due to severe rain storm
May 3	32	Accumulator No. 18 scrammed three drives
May 4 - 5	23	Control rod test pattern
May 11	74	Changed control rod pattern
May 11	134	Vertical shape test
May 12 - 13	34	Control rod worth test
May 15	34	In-core calibration
May 16	31	Repair accumulator No. 16
May 18 - 19	18	Control rod test pattern
May 24	45	Feedwater heater tests
May 25 - 26	23	Control rod test pattern

TABLE II

LOAD RESTRICTIONS (Cont'd)

Date	Reduction in Electrical Capability - MWe	Condition
June 1 - 2	28	Control rod test pattern
June 3	22	System load requirement
June 4	20	System load requirement
June 8 - 9	35	Control rod test pattern
June 9 - 10	34	Control rod worth test
June 13	34	Leakage test on heater spill valves
June 15	42	Control rod test pattern
June 16	67	System load requirement
June 17	27	Repair accumulator No. 20
June 26 - 29	34	In-core stabilization and calibration
July 2	34	Repair J-6 Asco valve
July 4	19	System requirement
July 7	11	Control rod test pattern
July 8 - 9	34	Control rod worth test
July 14	30	Control rod test pattern
July 17 - 23	19	Tube leak "B" Sec Stm. Gen.
July 27 - 29	19	Tube leak "B" Sec. Stm. Gen.
July 31	38	In-core calibration
August 1	25	Max. steam flow test
August 11	38	Control rod worth test
August 21	35	Repair scram outlet valve - accumulator No. 5
August 23	58	Press. regulator repair
August 24	28	Repair accumulator No. 23
August 26	36	Press. regulator oscillation
August 28	38	In-core calibration
August 29	28	Pressure regulator work
September 7	38	Pressure regulator work
September 9	38	Pressure regulator work
September 11	38	In-core calibration
September 14	38	In-core calibration
September 17 - 30	1-10	Fuel depletion
September 20	34	Press. regulator work and control rod worth tests
September 21 - 22	34	Control rod worth tests
September 25 - 27	30	Dual cycle mapping tests

TABLE II

LOAD RESTRICTIONS (Cont'd)

Date	Reduction in Electrical Capability - MWe	Condition
October 1 - 31	12-34	Fuel depletion
October 6	22	Control rod worth tests
October 8 - 9	23	Fuel leak test
October 12	22	Fuel leak test
October 18	18	In-core calibration
October 21	64	Tighten packing on "E" extraction non-return valve
October 23 - 31	5	Fuel leakage
October 24 - 28	13	Fuel leak test
November 1 - 7	36-38	Fuel depletion
November 1 - 7	5	Fuel leakage
November 5		In-core calibration
November 5 - 6	21	End of life nuclear test