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PATHFINDER ATOMIC POWER PLANT TECHNICAL PROGRESS REPORT

APRIL 1964 - JUNE 1964

Submitted to
**U. S. ATOMIC ENERGY COMMISSION
NORTHERN STATES POWER COMPANY**
and
CENTRAL UTILITIES ATOMIC POWER ASSOCIATES

by

**ALLIS-CHALMERS MANUFACTURING COMPANY
ATOMIC ENERGY DIVISION
Milwaukee 1, Wisconsin**

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Ref: AEC Contract No. AT(11-1)-589

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ALLIS-CHALMERS MANUFACTURING COMPANY

Under

Agreement dated 2nd Day of May 1957, as Amended
between

Allis-Chalmers Mfg. Co. & Northern States Power Co.
under

AEC Contract No. AT(11-1)-589

APRIL 20, 1965

Classification - UNCLASSIFIED

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PATHFINDER ATOMIC POWER PLANT

TECHNICAL PROGRESS REPORT

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CONTENTS

	<u>Page</u>
Distribution	1
List of Illustrations	3
Foreword	4
Design Data	5
6. Physics—Post-Construction R&D	6
6.1 Introduction	6
6.2 Test 302 - Pre-Experiment Analysis	14
References	19

ILLUSTRATIONS

<u>Fig. No.</u>	<u>Title</u>	<u>Page</u>
1	Pathfinder Boiler Core Slab Test 301 Configurations	7
2	Integral Rod Worth Curve of Central Control Rod in 15 and 18 Element Slab Cores	12
3	Differential Rod Worth Curve 16 Control Rod Bank Boiler Core	17
4	Axial Thermal Flux Distribution Boiler Core - All Rods Banked at 16.5 Inches	18

TABLES

<u>Table No.</u>	<u>Title</u>	<u>Page</u>
1	Measured k_{eff} for Slab Cores.	6
2	Post Experiment Calculated k_{eff} for Boiler Slab Cores	9
3	Zirconium Resonance Integral Data	11
4	Calculated Rod Worth for Central Control Rod in Slab Cores	13
5	Experimental vs. Calculated Slab Core k_{eff}	13
6	Experimental vs. Calculated Differential Rod Worth Results	13
7	Full Boiler Core Calculated Shim and Control-Rod Worth	15
8	Full Boiler Core Calculated Shutdown Margin	16

FOREWORD

This is one of a series of reports covering technical progress on the research and development program being performed in connection with the design of the Pathfinder Atomic Power Plant. This plant is located at a site near Sioux Falls, South Dakota. The Northern States Power Company of Minneapolis, Minnesota, is the owner and operator of the plant.

The U. S. Atomic Energy Commission, through Contract No. AT(11-1)-589 with Northern States Power Company, and Central Utilities Atomic Power Associates* (CUAPA), are sponsors of the research and development program.

Allis-Chalmers Manufacturing Company of Milwaukee, Wisconsin, under contract with Northern States Power Company, is performing the research, development, and design; and is constructing the plant including the reactor, which is designated the Controlled Recirculation Boiling Reactor (CRBR) with Nuclear Superheater. Pioneer Service and Engineering Company of Chicago, Illinois is providing the architect-engineer services to Allis-Chalmers. Portions of the R & D program, particularly in connection with fuel development, have been subcontracted by Allis-Chalmers.

*CUAPA MEMBER COMPANIES:

Interstate Power Company
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St. Joseph Light & Power Co.
Western Power and Gas Company
Wisconsin Public Service Corp.

DESIGN DATA

CRBR WITH NUCLEAR SUPERHEATER

Plant

Power, boiler region	157,200 kw
Power, superheater region	42,400 kw
Steam flow at rated power	616,125 lbs/hr
Total core power	199,600 kw
Gross electrical capability	66,000 kw
Net electrical output	62,000 kw
Net efficiency	31.0 per cent
Steam outlet pressure (reactor)	535 psig
Reactor operating pressure	600 psig
Temperature, boiler region	489 F
Outlet temperature, superheater region	825 F
Gross heat rate	10,199 Btu/kw-hr
Reactor building size	50 ft dia x 120 ft

Reactor

Vessel size (over-all)	11 ft 6 in o.d. 36 ft 1 in
Total core dimensions	6 ft x 6 ft
Dimensions of superheater region	6 ft x 30 in
Fuel, boiler (Zr-2 clad)	approx. 2.2 per cent enriched UO ₂
Fuel, superheater (S.S. clad)	approx. 93 per cent enriched UO ₂
Fuel, loading, boiler (U-235)	145.6 kg
Fuel, loading, superheater (U-235)	42 kg
Power density (boiler core coolant)	87 kw/liter
Average heat flux, boiler region	128,000 Btu/hr-ft ²
Average heat flux, superheater region	77,800 Btu/hr-ft ²
Maximum heat flux boiler region	462,000 Btu/hr-ft ²
Maximum heat flux, superheater region	245,000 Btu/hr-ft ²
Recirculation rate	65,000 gpm
Recirculation pump power	823 kw
Neutron flux	approx. 5×10^{13} n/cm ² sec

6. PHYSICS — POST-CONSTRUCTION R&D

6.1 INTRODUCTION

Test 301 (Boiler Slab Core Loading and Criticality) has been completed. This test was performed to provide information on the validity of the nuclear design model used for Pathfinder and to approach the Pathfinder full core loading in a safe manner. The test determined the minimum critical number of reference 2.2 w/o U-235-enriched fuel assemblies in a rectangular slab array. In addition, information was obtained on the reactivity worth of additional boiler fuel assemblies. This series of tests was performed for experimental determinations and for training operating personnel prior to loading the full core boiler and superheater.

This report presents the post-experiment analysis. This report also describes the pre-experiment analytical results for Test 302 (Full Boiler Core Loading and Criticality). The last quarterly report (ACNP-64024) presented the experimental results of Test 301. ACNP-64005 gave the pre-experiment analysis of Test 301. A topical report (ACNP-64601) covering Test 301 describes the pre- and post-experimental results in detail.

As described in ACNP-64024, initial criticality was achieved at Pathfinder with 15 2.2 w/o U-235 boiler fuel elements in a 3x5 slab array. Subsequently, the core was enlarged to 16 and 18 elements (Fig. 1). Table 1 lists the measured k_{eff} values for those three slab cores.

TABLE 1
MEASURED k_{eff} FOR SLAB CORES

<u>no. of boiler elements in slab</u>	<u>k_{eff}</u>
15	1.002 + 0.001
16	1.006 + 0.002
18	1.015 + 0.003

The Δk uncertainties in Table 1 are based on reproducibility of critical-rod heights and differential rod worths.

PATHFINDER BOILER CORE SLAB
TEST 301 CONFIGURATIONS

FIG. 1

Preliminary calculation had predicted criticality for a slab core consisting of 12 boiler fuel elements. Section 6.0.1 of ACNP-64005 describes this analysis, which is based on an iteration between XY and RZ eigenvalue problems. Results of the analysis are given on p. 27 of that same report.

Using the exact experimental conditions encountered during Test 301, an analysis based on the same model predicted criticality on 13 elements with $k=1.002$. This analysis also predicted a $k=1.015$ for a 15-element slab. Criticality was achieved on a 15 boiler fuel element slab array with a measured $k=1.002$. Thus, the additional two elements required for criticality represented a calculated loss in core reactivity of 1.3 percent Δk . To understand this loss, a post-experiment analysis was conducted for the slab core loading.

Several areas of uncertainty, some of which apply principally to the slab or minimum critical core loading, exist in the calculational model used in the Pathfinder reactivity analysis. These areas of uncertainty are outlined briefly in the following.

(1) Homogenization of Axial Gap Regions—The boiler core fuel assembly is divided into four axial fuel sections, separated by a nonfueled gap that includes a fission gas space. Located adjacent to these nonfueled regions are 1/2-in. buffer pellets containing reduced enrichment and a burnable poison, Dy_2O_3 , to suppress flux peaking. The homogenization for these regions is described in detail in Ref. 1.

The boiler fuel used to mock up the full Pathfinder core in the Allis-Chalmers critical facility did not contain axial gap regions as large as $UO_2 - Dy_2O_3$ buffer pellets. For this reason, the treatment for gap representation could not be verified at the critical facility.

No additional calculations were performed to investigate this uncertainty. However, a grid to maintain fuel-pin spacing within the boiler assembly is located at each of these gap regions. This grid previously had been represented in the calculations as Zr-11, but actually it is stainless steel. Each grid contains approximately 80 g of metal and is located at the center of the gap region in the moderator where the thermal flux is peaked. A calculation was performed to evaluate the effect of this stainless steel.

(2) Zirconium Resonance Integral—The boiler fuel elements in the Allis-Chalmers critical facility used aluminum as the structure and clad material, whereas Pathfinder uses Zr-11. The resonance integral for zirconium has been measured to be from 0.7 to 3.7 barns.² Because of the amount of structure material present in the boiler fuel assembly, this variation represents a significant uncertainty in calculated core reactivity.

The calculations for these tests used zirconium cross section data as compiled in the MUFT-IV library tape 5. Additional calculations to evaluate this uncertainty have not been performed, but a discussion of the reactivity worth of this variation is included in the results section.

(3) Multigroup Calculations—The preliminary calculations primarily used a two-group theory corrected by a four-group $\Delta\lambda$ (≈ 0.006). Additional calculations to evaluate the leakage from the small slab cores were performed using three- and four-group calculations. No new effect was found.

(4) Homogenization of Fuel-Rod Region, Zr Box, and Control-Rod Channels—In the preliminary analysis, the control-rod channels were explicitly represented, but the Zr box and water between fuel boxes had been homogenized with the fuel-rod region using flux weighting factors from constant source XY-PDQ cell calculations. Because of the small core size with large flux gradients, additional slab core detailed calculations were performed with the box and water represented explicitly.

In addition to the uncertainties present in the calculational model, some of the assumptions made in the preliminary calculations were invalid. The slab core had been assumed to be entirely surrounded by water-filled boiler boxes. The control rods, assumed to be entirely removed from the upper reflector region, were within 1/2 in. of the core in the upper reflector. Also, to have Channels 5 and 6 of the in-core instrumentation on scale before 1 and 2 saturated, it was necessary to locate them immediately adjacent to the core in the fuel assembly locations shown in Fig. 1. Part of the water reflector at that location was displaced. The reactivity worth of these effects was approximated.

Calculations evaluating these effects for the 15- and 18-element slab configurations were performed. A summary of these calculations and the reactivity effect of the various refinements are shown in Table 2.

TABLE 2
POST-EXPERIMENT CALCULATED k_{eff}
FOR BOILER SLAB CORES

<u>type of calculation and refinement</u>	<u>15-element slab core</u>	<u>18-element slab core</u>
(1) preliminary Model-two groups*	1.027	1.041
(2) same as (1) with four groups		1.036

*These k values are consistent with those listed in Table 11 of ACNP-64005.

Table 2 (continued)

<u>type of calculation and refinement</u>	<u>15-element slab core</u>	<u>18-element slab core</u>
(3) four-group and explicit fuel and box representation	1.009	1.022
(4) four-group, explicit fuel and box, s.s. grid at gap	1.005	1.018
(5) same as (4) with approximated worth of control rods and instrumentation in reflector	1.003	1.016
(6) same as (5) with three groups	1.004	--

Calculations evaluating the above were performed on the lower core in an XY plane. Approximately the same effect should be obtained in the upper boiler core region, so that the weighting between lower and upper cores is not changed. A neutron-balance calculation showed that the reactivity difference in the four-group calculations results from increased core leakage from the fast groups. For the explicit representation, the difference results from increased absorptions in the control-rod channel and box and water regions with a small increase in core leakage.

The worth of the stainless-steel grid in the axial-gap region was obtained from an RZ-PDQ calculation with the gaps represented explicitly. The homogenization of the grid in the gap region was performed using local P-3 transport calculations available from the reference core work.

The worth of the control rods in the upper-reflector region was approximated by extrapolation of a calculated control-rod worth curve. The worth of the instrumentation adjacent to the core was approximated by a ratio of unreflected core surface area.

A summary of the zirconium resonance integral data used in these calculations and the effect on the lower core k^∞ of variations in the resonance integral are shown in Table 3. For all slab core calculations, fast group constants, obtained from the MUFT IV code with the Tape-5 library, were used.³ In Table 3, there are three values for the zirconium resonance integral using the Tape-5 library. For these calculations, the unshielded value was used. The shielding referenced here represents the competition between the scattering and absorption within the various zirconium resonances.⁴ This condition cannot be treated explicitly within the coarse group description for the MUFT code and, consequently, is treated by using a self-shielding factor. No spatial self-shielding was assumed.

TABLE 3

ZIRCONIUM RESONANCE INTEGRAL DATA*

<u>source of data</u>	<u>Zr σ (R1)</u>	<u>k^∞</u>
MUFT IV Tape-0 Library (unshielded)	0.04 barns	1.302
MUFT IV Tape-5 Library (unshielded)	1.05 barns	1.280
MUFT IV Tape-5 Library (shielded)	0.85 barns	1.283
MUFT IV Tape-5 Library (infinite dilute).	1.17 barns	1.278

*MUFT calculations are for lower boiler core.

The change in reactivity resulting from a change of one barn in the resonance integral is significant. Based on the uncertainty in the measured value for the infinite dilute integral (0.7 to 3.7 barns), there is an appreciable reactivity uncertainty as a result of nuclear constant.

In addition to the integral control-rod worths calculated for the preliminary analysis, (i.e., the worth of control rods 12, 15, and 16 in the 12-element slab, (Table 11, ACNP-64005), the integral and differential worth of the central control rod 12 were calculated for both the 15- and 18-element slab configurations. The integral worth was computed using explicit XY-PDQ calculations with control-rod constants as used in the reference core analysis. The differential worth was obtained by performing Z dimensional windowshade calculations with the WANDA-4 code.

In determining the differential rod worth, group dependent radial bucklings from the explicit XY-PDQ calculations for the control-rod-out and rod-in conditions were used in their respective axial regions. The control-rod absorptions were simulated by a homogeneous thermal poison in the core. The specific amount of control poison was obtained from a poison search matched to the XY calculation of rod worth. Before performing the calculation of rod worth, the excess reactivity was matched to the experimental value by a homogeneous fast poison. The results of these calculations are summarized in Table 4 and Fig. 2.

INTEGRAL ROD WORTH CURVE OF CENTRAL CONTROL ROD
IN 15 AND 18 ELEMENT SLAB CORES

FIG. 2

TABLE 4

CALCULATED ROD WORTH FOR CENTRAL
CONTROL ROD IN SLAB CORES

<u>number of elements in slab</u>	<u>control-rod worth 12, percent Δk (all other rods out)</u>
12	9.1
15	7.6
18	6.1

Results from the post-experiment calculational analysis are in good agreement with the experimental results. A summary of the comparison between experimental and calculational results comparing the core excess reactivity and control-rod worth data is shown in Tables 5 and 6.

TABLE 5

EXPERIMENTAL vs. CALCULATED
SLAB CORE k_{eff}

<u>number of elements in slab</u>	<u>experimental k_{eff}</u>	<u>calculated k_{eff}</u>
15	1.002 ± 0.001	1.003
18	1.015 ± 0.003	1.016

TABLE 6

EXPERIMENTAL vs. CALCULATED
DIFFERENTIAL ROD WORTH RESULTS

<u>number of elements in slab</u>	<u>experimental $\frac{\partial \rho}{\partial H}$ (%Δk/in.)</u>	<u>calculated $\frac{\partial \rho}{\partial H}$ (%Δk/in.)</u>
15	0.032	0.044
16	0.055	0.048
18	0.090	0.090

For these slab arrays, three neutron groups are adequate. The explicit representation of geometric detail in k_{eff} calculations, as opposed to using cell theory, is significant; and, of course, careful attention to the actual experimental conditions is important. However, large areas of reactivity uncertainty (such as in the zirconium resonance integral data) are still present in the calculational results and some compensating error effects are probably present in these calculations.

6.2 TEST 302—PRE-EXPERIMENT ANALYSIS

Test 302 (Full Boiler Core Loading and Criticality) is scheduled to follow Test 301. The objective of Test 302 is to load, in a safe manner, a complete boiler core with 96 elements enriched to 2.2 w/o U-235. Critical experiments with this core will provide a basis for verifying the analytical methods used to compute criticality of the full Pathfinder core. In Test 303, the superheater core is added to complete the full core loading. Comparison between Tests 302 and 303 results will give the superheater worth as well as test the analytical methods used to compute the superheater nuclear characteristics. Thus, Tests 301, 302, and 303, considered as a whole, will provide criticality data for a minimum critical mass of boiler fuel, full boiler core, boiler plus superheater core, and the superheater region alone.

The general test method of 302 follows that of 301. The final 301 core will be in position, and the water level will be below the fuel region with all control rods full in. Thirty-two boron steel poison shims will be installed in the core. Boiler elements will be added, one at a time, to the dry slab core until the boiler core is fully loaded (96 2.2 w/o elements). The reactor head will then be installed, control rods worth about 4 percent Δk will be partially withdrawn and cocked, and water will be added in predetermined increments. A conservative analysis which assumes that no poison shims are in the core and that all control rods are full out predicts that criticality on water height could be expected at 18 in. of water. Thus, below 18 in. of flooded core height, water fill increments will be taken to provide sufficient data points for an accurate prediction of critical water height, if criticality is to be imminent. Water fill increments will be taken above 18 in. to limit the reactivity addition to less than 3 percent $\Delta k/k$, which is less than the 4 percent $\Delta k/k$ in cocked rods. Also above 18 in., the rate of reactivity addition will be limited to less than 2 ϵ /sec. Water will be added at the rate of either 30 or 60 gpm. These additions correspond to a rate of flooded core height addition of 0.01 in./sec or 0.02 in./sec, respectively.

Count rates will be taken on the in-vessel chambers after the addition of each element or each water increment. Inverse count rate data will be plotted and then extrapolated to check for criticality on the next incremental addition. Criticality will not be permitted during the fuel loading or water fill.

The boron steel poison shims will be removed in groups of eight to provide data for five boiler cores. Each core will be more reactive than the preceding core. For each core, the shutdown margin with all rods in and with the most reactive rod full out will be measured. Critical rod positions for a 16-rod bank and for each of three control-rod groups will be measured. At each critical position, a differential rod worth measurement will be made. The removal of the poison shims will be limited so that the safety margin will always be met with the most reactive rod full out.

The methods used in the Test 302 analysis are exactly the same as in the final analysis of Test 301 - three neutron groups, detailed geometric representation in the XY plane, and the axial buckling determined from an RZ calculation (see ACNP-64601, Secs. 3.1 and 3.2). Based on these methods, the following core k values were calculated.

TABLE 6

TEST 302 FULL BOILER CORE CALCULATED k VALUES

<u>core condition</u>	<u>k</u>
All rods and all shims out	1.076
All rods out, 32 shims in	1.056
16 rods in, all shims out	0.908
16 rods in, 32 shims in	0.883

The worth of the boiler control rods and the 32 poison shims are listed in Table 7.

TABLE 7

FULL BOILER CORE CALCULATED SHIM AND
CONTROL-ROD WORTH

<u>core condition</u>	<u>%Δk</u>
32 shims, rods out	2.0
32 shims, rods in	2.5
16 rods in, shims out	16.8
16 rods, shims in	17.3

The worth of the most reactive rod with the other 15 rods full in and 32 shims in was calculated to be 6.3 percent Δk . With no shims in, this worth is estimated to be 6.6 percent Δk . The core shutdown margin as a function of core condition is summarized in Table 8.

TABLE 8

FULL BOILER CORE CALCULATED SHUTDOWN MARGIN

<u>core conditions</u>	<u>%Δk</u>
16 rods in, 32 shims in	11.7
15 rods in, 32 shims in	5.4
16 rods in, all shims out	9.2
15 rods in, all shims out	2.6

Thus, for the shim free boiler core with the most reactive rod full out, the shutdown margin is predicted to be 2.6 percent Δk .

For each core, the initial approach to criticality will be made with all 16 boiler control rods in a bank. For the initial 32 shim boiler core, the expected 16-rod bank critical position is 16 in. This prediction is based upon a synthesis of XY all-rods-out and all-rods-in PDQ's using a WANDA calculation along the Z axis. Flux weighted constants for the boiler core were obtained from the PDQ's for input into the WANDA calculations. The WANDA problems were normalized to the all rods out k from PDQ by adjusting the transverse buckling (XY plane buckling) in WANDA. A homogeneous poison was calculated to represent the control-rod bank in the WANDA problems so that the total bank worth in WANDA is the same as in PDQ. These values of k and rod worth are given in Tables 6 and 7 as 1.056 and 0.173, respectively. With this problem description, a 16-rod bank search was performed to find the bank critical height. Based on this one-dimensional synthesis, 14 in. was predicted for the critical 16-rod bank position. To confirm this expected critical position but without the uncertainty resulting from the transverse buckling approximation, an RZ problem was run with the bank at 14 in. This problem was slightly subcritical. Using the differential rod worth curve shown on Fig. 3, it was estimated that the RZ problem would have been critical if the rod bank had been at 16 in. Thus, the expected critical position of the 16-rod bank in the initial Pathfinder boiler core is 16 in.

The Fig. 3 curve was generated using a series of WANDA's. Figure 4 shows an axial flux plot as calculated using WANDA with the rod bank at 16.5 in.

DIFFERENTIAL ROD WORTH CURVE,
16 CONTROL ROD BANK BOILER CORE

FIG. 3

AXIAL THERMAL FLUX DISTRIBUTION BOILER
CORE - ALL RODS BANKED AT 16.5 INCHES

FIG. 4

REFERENCES

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2. Neutron Resonance Integral and Age Data, Spinrad et al, ANL Newsletter #1
3. 54-Group Library for P-1 Programs, A. F. Henry, WAPO-TM-224
4. Reactor Technology Report 23 - Physics, KAPL 2000-20, May 1963