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PATHFINDER ATOMIC POWER PLANT
SUPERHEATER FUEL THERMOCOUPLE
CALIBRATION TESTS (711A)
AND
RADIATION COOLING TESTS (723)

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
Bethesda, Maryland

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

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AND

RADIATION COOLING TESTS (723)

by L. L. Kintner, G. S. Green, Jr., J. P. Kelly, and L. B. Myers

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ALLIS-CHALMERS MANUFACTURING COMPANY

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PATHFINDER ATOMIC POWER PLANT

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AND
RADIATION COOLING TESTS (723)DISTRIBUTION

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FOREWORD

One of a series of reports on research and development in connection with the design of the Pathfinder Atomic Power Plant, this particular report deals with the Superheater Fuel Thermocouple Calibration Tests (711A) and the Radiation Cooling Tests (723). The Pathfinder Plant is located at a site near Sioux Falls, South Dakota and reached criticality early in 1964. Owners and operators of the plant will be Northern States Power Company of Minneapolis, Minnesota. Allis-Chalmers is performing the research, development and design, as well as being responsible for plant construction.

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. The plant's reactor will be of the Controlled Recirculation Boiling Reactor type with Nuclear Superheater.

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1. INTRODUCTION

The superheater fuel elements for the Pathfinder Reactor are designed to be cooled by radiation and conduction of fission product decay heat to the superheater moderator (water) a short time after the reactor shutdown with no steam flow through the superheater. Each superheater element contains an unfueled central rod and two concentric cermet fuel tubes. Each element is located in a channel which is formed by an insulating tube and a pressure tube. Moderator water flows outside the pressure tube. During radiation cooling, heat is transferred across the steam annuli to the moderator water. The superheater moderator water is recirculated by the boiler recirculation pumps since the superheater moderator space is in a parallel flow path with the boiler core. Heat is removed from the reactor water (if necessary to maintain or decrease pressure) by blowing down reactor water through the purification system and returning colder water to the reactor through the reactor gland seal system.

Decay heat immediately after shutdown is removed by steam flow through the superheater. Steam flow is stopped when the fission product heat has decayed to a level such that radiation and conduction of heat to the moderator will result in superheater temperatures less than the design upper limit of 1300 F. It is desirable to predict steam flow cut-off time as accurately as possible because excessive steam blowdown may cause excessive water temperature decreases. Water temperature decreases after shutdown are limited by reactor vessel stress considerations. Low water temperature after shutdown may also be significant to power plant economics, since it would require a longer time to reestablish power generation.

Superheater temperatures with radiation cooling have been calculated and measured in an out-of-pile experiment during the design phase (see Ref. 1). Reactor shutdown procedures are based upon the results of out-of-pile testing and design calculations. The purpose of the radiation cooling tests in the reactor was to gain added assurance that maximum superheater temperatures during radiation cooling are predicted by design calculations. The tests were performed early in core life, using fission heat to simulate fission product decay heat, so that heating could be controlled. Forty-five thermocouples were attached to five instrumented fuel elements to measure superheater temperature. Thermocouple calibration was checked prior to the test by adiabatic runs at several reactor water temperatures. The results are compared with design calculations; the accuracy of the design calculations is assessed; and the method and data for computing peak superheater temperatures after reactor shutdown are summarized.

2. SUMMARY AND CONCLUSIONS

2.1 ACCURACY OF CALCULATED SUPERHEATER TEMPERATURES

The accuracy of the heat transfer and neutron flux distribution calculations is evaluated by comparing the linear heat rate Q'_m computed from measured temperatures with the calculated linear heat rate, Q'_c . Linear heat rate is the heat transfer per unit length

from one fuel element, Btu/(hr)(ft). A value of $Q_m^i/Q_c^i = 1.0$ is obtained when measured temperature equals calculated temperature. Values of Q_m^i/Q_c^i for each thermocouple and each run of the test are given in Table 4. Values near the peak flux regions (at 39 and 57 in. from the bottom in the four outer elements) are summarized below.

Fuel Element Component	Q_m^i/Q_c^i		
	Minimum	Average of 8 Runs	Maximum
Poison Tube	0.78	0.91	0.99
Inner Fuel Tube	0.74	0.91	1.04
Outer Fuel Tube	0.89	1.07	1.23

It is concluded that the design calculational models of heat transfer and neutron flux distribution predict superheater fuel temperatures near peak flux regions within the accuracy of the calculations. The expected range of Q_m^i/Q_c^i for the test was 0.75 to 1.33. The maximum temperatures, which occur on the poison and inner fuel tube, are slightly overpredicted and the outer fuel tube temperatures are slightly underpredicted, as seen in the above tabulation. This result may also be seen in the temperature plots of Figs. 32, 33, and 34 for thermocouples at 39 and 57 in. The underprediction of the outer fuel tube temperature indicates a slight inaccuracy in the heat transfer model, since the neutron flux level at a constant elevation in an element is approximately the same for all three tubes. Prediction of outer fuel tube temperature is not as significant as prediction of poison and inner fuel tube temperature since the maximum temperatures, which are of prime importance, always occur on the poison and inner fuel tube during radiation cooling.

In the regions of lower neutron flux, the calculational models overpredict superheater temperatures by a significant amount. Measured temperatures at 21 in. from the bottom of the outer elements and in the control element are significantly lower than calculated, as shown in Figs. 13, 14, 27, 28, 30, and 31. This overprediction is probably due to overprediction of neutron flux level in the lower part of the core and in the core center. This observation is consistent with previous flux mapping tests and subsequent power operation.

The design calculational model can be used to calculate peak superheater temperatures up to 1300 F within the accuracy of the calculations (see Sec. 2.2). Figures 32 and 33 show that the maximum measured temperature as a function of power closely follows the shape of the calculated curve over the range of measured temperatures (500 to 740 F).

2.2. METHOD FOR CALCULATING PEAK SUPERHEATER TEMPERATURES DURING SHUTDOWN

Design data for calculating peak superheater temperatures during radiation cooling with the reactor shut down are summarized in Sec. 7. The heat transfer model is the same as that used in design calculations (see Ref. 1) but latest refinements in flux distributions

have been incorporated. The upper limit of temperature is calculated by applying an uncertainty factor of 1.54 on the most probable calculated linear heat rate. This uncertainty in linear heat rate is larger than the maximum ratio of measured to calculated heat rate (1.23) due to the larger uncertainty in predicting linear heat rate for shutdown powers, as discussed in Sec. 7.3.

2.3 SUPERHEATER THERMOCOUPLES

Superheater thermocouples which have continuous leads in the reactor are more reliable than those having connectors in the reactor. Readings obtained from thermocouples with connectors varied considerably from reactor water temperature during adiabatic runs, when the connectors were under water. When connectors were in steam, the measured temperatures were much closer to expected temperatures during adiabatic runs. The continuous lead thermocouples were always within ± 5 F of reactor water temperature.

The number of working thermocouples at the start of the radiation cooling test was 25 out of a total of 45 installed. The working thermocouples were satisfactorily determined in adiabatic runs prior to the test. No unexpected thermocouple behavior was observed during the test.

Each ice junction for the thermocouple recorders should be in a separate, insulated container in the ice bath. Some drift of temperature readings and noise on readouts was observed prior to putting ice junctions in separate insulated containers.

3. TEST PROCEDURE

After installation of the instrumented fuel elements, a test was run to determine working thermocouples (Test 711A; see Ref. 9). Readout of all thermocouples was compared to reactor water temperature measurements during adiabatic runs at several temperature levels up to 480 F. The superheater was flooded during these runs and reactor water was recirculated through superheater elements.

For the radiation cooling test (Test 723), the reactor water was heated to 420 F with the superheater flooded. The superheater was drained and the reactor was made critical at a low power level. Power was increased in small increments which resulted in maximum temperature increases of 50 F in the superheater fuel elements. Reactor water temperature was maintained at about 420 F by blowing down water from the reactor through the purification system while adding water at about 90 F through the reactor gland seal system. The detailed test procedure is given in Ref. 2.

The control rod configuration for the test was selected to have a large fraction of reactor power in the superheater to simulate fission product heating after shutdown from full power. Control rod Groups I and II were withdrawn, Groups IV and V were inserted, and Group III was controlling (see Fig. 1). The critical position of Group III rods was about 26 in. with reactor water temperature of 420 F.

Superheater temperatures were measured at power levels of 0.18, 0.32, 0.49, 0.64, 0.79, and 0.97 Mwt during power escalation with two additional runs (0.97 and 0.46 Mwt) to check reproducibility. At each power level, equilibrium conditions were achieved. Approximately 45 min were required to reach equilibrium after a change in power level. Several runs longer than 1 hr were made and in two of the runs (0.64 and 0.46 Mwt) the equilibrium condition was approached from a higher power level.

The reactor power was determined from Channels 5 and 6 ion chambers, which had been previously calibrated at 2.34 Mw and 420 F reactor water temperature for this rod configuration (see Ref. 3). During the 0.97 Mwt run, a heat balance was made to check the calibration of Ref. 3. The water flow into the reactor was measured by shutting off water blowdown and timing the level change in the reactor and in the condenser hot well. This heat balance gave a calculated power level of 1.08 Mwt compared to 0.97 Mwt from the previous calibration of Ref. 3.

The reactor power for the test was limited to less than 1.2 Mwt which limited expected peak superheater temperatures (including uncertainties) to less than 1000 F. This power and temperature was believed to be sufficiently high to evaluate the heat transfer calculational model and permit extrapolation to design temperature (1300 F) with sufficient accuracy.

Safety circuits for the test included a high level scram on nuclear Channels 5 and 6 at 1.5 Mwt and a variable setpoint runback on nuclear Channel 4. The runback was set slightly higher than the desired equilibrium power level for each run during power escalation. Runbacks occurred several times during the test and the protective system performed its function well. Runback lowered power slightly by a slight insertion of control rods. Scram did not occur during the test.

Reactor water temperature was measured by thermocouples in the discharge pipe from all three recirculation pumps. The three measurements were within 2 F of each other. The average of the three water temperature measurements was 6 F less than saturation temperature corresponding to measured reactor dome pressure.

The superheater fuel element temperature measurements, which are of prime interest for this test, are described in Sec. 4.

4. SUPERHEATER TEMPERATURE MEASUREMENTS

4.1 DESCRIPTION OF THERMOCOUPLES

There are five instrumented fuel elements in the superheater core (see Fig. 1). Four are in locations having the highest heat generation and one is in the center of the core. Two of the elements have thermocouple hot junctions located primarily to measure temperatures during radiation cooling (Type R) when peak temperatures are expected to occur near the core vertical center. Three of the elements have hot junctions located primarily to measure temperatures during steam cooling (Type S) when peak temperatures are expected to occur near the outlet end of the fuel elements.

The specific location of hot junctions in the two types of instrumented superheater elements and the orientation of the elements in the core are shown in the fuel element cross sections of Figs. 2 and 3. The instrumented fuel elements (see Refs. 5 and 6) have the same dimensions as normal fuel elements (see Ref. 4). The fuel is contained in the two concentric tubes and poison is contained in a stainless steel tube in the center of the element. The central poison rod, inner fuel tube, and outer fuel tube are held in an approximately concentric position by nine spacer wires in a normal fuel element. In an instrumented fuel element, nine thermocouples are used in place of the nine spacer wires. The fuel elements are located in channels formed by the pressure tube and an insulating tube inside the pressure tube. The annular space between the pressure tube and the insulating tube is vented to the steam environment to provide an essentially stagnant steam gap which prevents excessive heat loss from superheater steam to the water outside the pressure tube during power operation.

The instrumented elements which are designed for radiation cooling tests (see Fig. 2) have three hot junctions near the vertical center, three 18 in. above the center and three 18 in. below the center. At each elevation, one thermocouple is welded to the poison tube, one is welded to the inner fuel tube and one is welded to the outer fuel tube. All three thermocouples at a given elevation are within a 60-deg segment of the fuel element; two are in the same axial plane. This arrangement was selected as the best arrangement for evaluation of temperatures with radial heat flow, while maintaining proper spacing of the tubes. The evaluation of the radial heat flow calculational model is more meaningful if thermocouple hot junctions at a given elevation are close to each other, since uncertainties in the calculated flux gradient across an element can be minimized. Comparison of the measured and calculated temperature level at the three elevations in a Type R element permits an evaluation of the overall accuracy of the heat flux distribution and temperature calculational models. Thermocouples at the upper and lower elevations permit detection of a skewed axial heat flux distribution.

The instrumented elements which are designed principally for measuring temperatures with steam flowing through the elements (see Fig. 3) have six thermocouples near the exit end of the element and three near the center. The three in the center are most useful for radiation cooling tests. Of the six near the outlet ends, five hot junctions are located in the lower foot of fuel element where peak temperatures are expected during power operation. One (I-00) is located below the fuel element to measure the steam leaving all three annuli.

The thermocouples are 0.012-in.-dia chromel-alumel wires insulated with Al_2O_3 , and are encased in a 0.009-in.-thick stainless-steel sheath. The poison and outer fuel thermocouples are 0.044-in. OD and the inner fuel thermocouples are 0.068-in. OD. The hot junctions are ungrounded. The thermocouple sheath extends the full length of the element to serve as a spacer, but the portion below the hot junction contains only Al_2O_3 .

The thermocouples are spot welded to the outer surface of the fuel tubes and the poison tube. The diameter of the spot weld is 7 to 8 mils and the spacing between welds is 25 to 28 mils. Photomicrographs of cross sections through typical welds on the poison tube, inner fuel tube, and outer fuel tube are shown in Figs. 4, 5, and 6.

The response of some of the thermocouples to a sudden change of environment temperature (room temperature to boiling water) was measured prior to attachment to the fuel elements. The 0.068-in.-dia thermocouples have a time constant of about 0.5 sec and the 0.044-in.-dia thermocouples have a time constant of about 0.1 sec.

The measured hot junction temperature is less than the surface temperature since hot junctions are ungrounded and sheaths are spot welded to the surface. Heat transfer from the surface, through the thermocouple, to the heat sink is similar to heat transfer through a fin. During radiation cooling, when surface heat transfer coefficients are low, the thermocouple temperature is close to the surface temperature. During power operation, when surface coefficients are high due to flowing steam, the thermocouple temperature is not as close to the temperature of the surface to which it is welded. The calculations of the junction temperature relative to the surface to which it is welded are described in Sec. 5.3.

Thermocouple leads are chromel-alumel. The leads are routed through a 0.5-in.-dia tube from a few inches above the fuel tubes to the top of the superheater channel (see Ref. 5). The additional pressure drop due to the leads and special end fitting at the top of the assembly is 1.1 psi for full power conditions. Since pressure drop in a normal fuel element at full power is 60.4 psi (see Ref. 1), the steam flow in the instrumented element is only about 1 percent less than that in a normal fuel element.

Two types of thermocouple leads are used inside the reactor. Continuous leads which have an integral sheath inside the reactor are used for elements A-18 and E-9 (see Ref. 7). Jumper leads, which have plug-type connectors (see Ref. 8) at each end, are used for elements W-9, Z-1 and K-10. The thermocouple leads are seal welded to the connectors, but the male and female contacts of the connector are exposed to the reactor environment. Initially, the thermocouples for all five instrumented elements had jumper leads. The advantage of jumper leads is that the jumpers can be removed for access to the boiler core without removing the instrumented superheater elements. The connectors performed satisfactorily in an autoclave test prior to installation. However, after Phase I tests in the reactor were completed, a check of thermocouple readout through the connectors showed inconsistent behavior. These erratic readings were believed to be caused by extraneous emf's generated by deposits on the contacts. Cleaning of the contacts did not remedy the erratic behavior. Therefore, two of the instrumented assemblies were designed for continuous leads inside the reactor to provide greater assurance of reliable readings.

Thermocouple leads are routed through holes in the superheater baffle, across the boiler core holddown assembly, and through the water column. Photographs of the leads after

installation are shown in Figs. 38a, 38b, 39, and 40. Figures 38a and 38b show the three jumper leads (lower part) and continuous leads (attached to the top of steam separators). Figure 39 shows the top end fittings of two instrumented elements. Figure 40 shows routing of leads above the superheater. Instrumented elements having continuous leads were installed, the leads were routed through the water column and then external leads were spliced to them. The internal leads are of sufficient length so that the instrumented elements can be removed from the superheater and placed in storage tubes in the region of the steam separators thus permitting removal of the boiler core hold-down assembly.

Two Barber-Coleman 24-point recorders are located inside the containment and three Leeds and Northrup continuous recorders are located in the control room. Each recorder has a separate chromel-alumel ice junction.

4.2 THERMOCOUPLE CALIBRATION CHECKS

Several types of checks were made before and between runs of the radiation cooling test to determine which thermocouples were functioning properly. Before installation of the instrumented fuel elements, the response of the thermocouples was measured as the instrumented elements were immersed in hot water. The resistance was measured in the leads for elements with connectors. After installation, Test 711A was run, in which reactor water was circulated through the superheater, and thermocouple temperatures were compared with reactor water temperature measurements. Comparisons of superheater temperature and reactor water temperature were also made when the superheater was drained and the reactor water was cooled down by dumping steam to the condenser. Between runs of Test 723, when fission and fission product heating was negligibly small, thermocouple temperatures were compared with reactor water temperature.

The results of the calibration test (711A) are given in Table 1. Working continuous-lead thermocouples (Elements E-9 and A-18) always gave readings within 5 F of reactor water temperature. Thermocouples with connectors inside of the reactor read considerably different from reactor water temperature and were not consistent during the equilibrium runs with the superheater flooded. During cooldown with the superheater drained, some of the thermocouples with connectors were consistently closer to reactor water temperature. It is postulated that deposits on the connector contacts cause much larger spurious emf's when wet (superheater flooded) than when they are dry (superheater drained). Since superheater thermocouples are used only with the superheater drained, some of the thermocouples with connectors are used in the evaluations. However, they are less reliable than continuous lead thermocouples.

A summary of all calibration checks up to the end of Test 723 and an evaluation of working thermocouples are given in Table 2. The calibration checks between runs of Test 723 (Col. 6 to 9), with few exceptions, confirmed the results of preceding checks. During the first part of the test (Col. 6, 7, 8), the ice junctions for the multipoint recorders were outside of the insulated container in the ice bath which caused many of the thermocouples to read high. The ice junctions for the continuous recorders were in separate

insulated containers in the ice bath and these recorders read within 2 F of a portable precision potentiometer. After putting the ice junction for the multipoint recorders in insulated containers (Col. 9), most of the working thermocouples were within 5 F of the water temperature. It was concluded that improper location of the ice junction in the ice bath was causing a drift in the thermocouple readout.

Test 723 data obtained prior to relocating the ice junction were corrected based on calibration checks made between runs of the test. The corrections applied to measured temperatures are as follows:

THERMOCOUPLE CORRECTIONS FOR DRIFT DURING TEST 723, °F

		Runs 1-6	Runs 7-8	Runs 9-10
<u>Core Location</u>	<u>Thermocouple No.</u>			
E-9	I-57	-45 F	0	0
K-10	{ I-39, O-39, P-21, I-21, O-21 P-08 I-04 (except Runs 1 and 2)	-30 F	-18 F	0
Z-1				
Z-1	I-39	-18 F	-20 F	0
W-9	{ I-39, O-39, O-10, O-06, I-04, I-00			

The thermocouples are rated in the last column of Table 2 as "good", "fair", "poor", or "failed". The thermocouples rated "fair" or "poor" are used in the evaluation but are not considered as reliable as continuous lead thermocouples which are rated "good". The performance of thermocouples during the runs was also used to rate them; i.e., whether readings wandered, or responded to power changes as expected. In general, performance during the runs agreed with expected performance from calibration checks.

5. CALCULATED TEMPERATURES

Calculation of thermocouple temperatures includes calculation of fuel element surface temperatures for radial heat flow through the element, calculation of heat generation distribution, and calculation of thermocouple hot junction temperature relative to fuel

surface temperature. These three parts of the calculation for conditions of Test 723 are described separately in the following sections.

5.1 HEAT TRANSFER

Fuel element temperatures are calculated by assuming radiation and conduction heat transfer across the four steam channels of the fuel element (see Fig. 2). The temperature drop across metallic components is negligibly small. The equation for heat flow across one annular steam gap is:

$$q' = q'_{\text{conduction}} + q'_{\text{radiation}}$$

$$q' = \frac{2 \pi k}{\ln(d_o/d_i)} (T_i - T_o) + \frac{\sigma \pi d_i (T_i^4 - T_o^4)}{\left[(1/\epsilon_i) + (d_i/d_o) (1/\epsilon_o - 1) \right]} \quad (1)$$

where q' = heat transferred across steam gap, $\frac{\text{Btu}}{\text{hr ft}}$

k = thermal conductivity of steam, $\text{Btu/hr ft } ^\circ\text{F}$

$= 0.003 + 0.00004 (t_m)$; t_m is the arithmetic mean of the annulus surfaces, $^\circ\text{F}$

d = annulus diameter, ft

T = absolute temperature, $^\circ\text{F}$

σ = Stefan - Boltzman constant

$= 0.1742 \times 10^{-8} \text{ Btu/hr ft}^2 \text{ R}^4$

ϵ = surface emissivity = 0.8

and subscripts i and o refer to the inner and outer surface of the steam annulus.

The heat transferred across each of the four steam gaps of the element depends on the distribution of heating in the poison pin, fuel tubes, and insulating tube. For fission heating such as in Test 723, the heat transfer across each steam gap is calculated as:

$$q'_p = 0.026 Q' \quad (2)$$

$$q'_i = 0.410 Q' \quad (3)$$

$$q'_o = 0.942 Q' \quad (4)$$

$$q'_s = 0.953 Q' \quad (5)$$

where Q' is the total heat generated per foot of fuel element, Btu/hr ft and subscripts p, i, o, and s, refer to poison tube, inner fuel, outer fuel, and stagnant steam annuli (see Fig. 2). The remaining fraction of the total heat (0.047) is generated in the pressure tube and in the moderator outside the pressure tube and does not contribute to superheater fuel temperatures during radiation cooling. The distribution of heating in the components of the element for fission product heating are different than for fission heating (see Sec. 7).

Fuel temperatures were calculated by the solution of equations (1) through (5) on an IBM-1620 computer. Results are plotted in Fig. 7, which gives fuel element temperature as a function of the total linear heat rate of the fuel element. The linear heat rate includes heat generated in the pressure tube and moderator as well as heat generated in the poison rod, fuel tubes, and inner insulating tube.

Conduction of heat through the steam is a significant fraction of the total heat flow. For high temperatures and large gaps, radiation heat transfer is larger; but for low temperatures and small gaps, conduction is larger. For example, consider the case in which the poison tube temperature is 1270 F. In the poison tube annulus, 62 percent of the total heat flow across the annulus is due to radiation and 38 percent is by conduction. In the stagnant steam annulus (which is the coldest annulus), only 32 percent of the heat flow is by radiation and 68 percent is by conduction.

Heat transfer through the thermocouples is calculated to be about 25 percent of the total heat transfer across the inner fuel and outer fuel annulus for fuel temperatures around 800 F. However, the emissivity of 0.8 used in the heat transfer model is probably high for stainless steel. The use of a high emissivity tends to offset the neglect of conduction through spacer wires in the heat transfer model. It was determined in Ref. 1 that the use of an emissivity of 0.8 in the simplified model (without considering spacer wires) best fits the out-of-pile experimental data for radiation from a simulated element. Therefore, the simplified model of equation (1) adequately described the heat transfer in the element.

The fuel element dimensions used in the heat transfer model (see Fig. 2) are nominal dimensions for an element in the hot, operating condition. The design analysis showed that the effect on maximum element temperature of using maximum steam annulus thickness is small. For example, Fig. 2.33 of Ref. 1 shows that for a particular heating rate, an inner fuel tube temperature of 1200 F is calculated using nominal dimensions; if maximum dimensions are used, the fuel temperature would be 20 F higher.

5.2 NEUTRON FLUX DISTRIBUTIONS

The distribution of heating in the superheater core was calculated for the Test 723 control rod configuration (see Ref. 2). The distribution is calculated in terms of superheater power fraction (Q_{sh}/Q_r), ratio of fuel element power to average fuel element power in the superheater (ϕ_r), and ratio of heat generation at a given axial position to the average heat generation in that element (ϕ_a). Of the five core locations which contain instrumented

fuel elements, location A-18 is symmetrical with location Z-1 and location E-9 is symmetrical with W-9. Location K-10 is near the core center.

The heat flux ratios, ϕ_a , ϕ_r , and ϕ_l for each working thermocouple during Test 723 are included in Table 3. The flux gradient across a fuel element (ϕ_l) has a small effect on fuel temperatures even in those elements near the superheater boiler interface. The element in the center of the core has much less flux gradient across it. The local flux at each thermocouple location in elements A-18, Z-1, W-9, and E-9 is included in the calculation of thermocouple temperatures for these elements. The maximum local flux is 1.07 times the average in an element.

The linear heat rate, Q^l , for a particular thermocouple location is

$$Q^l = \frac{Q_r (Q_{sh}/Q_r) \times 3.413 \times 10^6}{411 \times 6} \times \phi_r \times \phi_a \times \phi_l \quad (6)$$

where Q_r is total reactor power, Mwt, and other symbols are as defined above. For Test 723, there were 411 superheater fuel elements in the core; the active core length is 6 ft.

5.3 RELATIONSHIP OF THERMOCOUPLE TEMPERATURE TO FUEL TEMPERATURE

The thermocouple temperatures are less than the temperature of the tubes to which they are welded because of the method of attachment (see Sec. 4.1). The magnitude of this temperature difference depends on the surface heat transfer coefficient of the thermocouple. The magnitude of thermocouple temperature relative to the temperature of the tube to which it is welded has been calculated by two methods.

In one method of calculation, a digital computer code (HOC) was used assuming conservatively that the hot junction was located between two spot welds and that heat transfer through the welds would not influence hot junction temperature (see Ref. 2). The thermocouple temperature relative to tube temperatures from this calculation was approximately the arithmetic mean between the temperatures of the two surfaces of the annulus. This solution is conservative because consideration of heat transfer through the welds would result in a higher calculated hot junction temperature and performance during the test was evaluated by comparing measured and calculated thermocouple temperatures. For the radiation cooling test, the error in predicting maximum element temperature (poison tube) was small since only a small temperature difference was expected between poison tube and inner fuel tube. However, the error in predicting inner fuel and outer fuel thermocouple temperatures is significant (see Fig. 7). The preliminary evaluation of superheater performance during Test 723 (see Ref. 11) assumes the thermocouple temperature is the mean between the temperatures of the two surfaces of the annulus.

A more detailed calculation of hot junction temperature relative to tube temperature was made principally for power operation where larger temperature differences between fuel

element and bulk coolant were expected. In this analysis, the effect of thermocouple-tube welds on radial heat flow and on axial heat flow were considered. A two-dimensional paper analog was used to determine temperature distribution in a cross section through a weld and in a cross section through a nonwelded section. From these solutions, the thermal resistance between the thermocouple junction and the tube was obtained for a welded section and a nonwelded section. These resistances for cross flow of heat were combined with the resistances for axial heat flow in the fuel tube and thermocouple in a model to determine the combined effect. Two values of fuel channel heat transfer coefficient (320 and 640 Btu/hr ft² °F) and two values of the "effective coefficient" between thermocouple and fuel tube of a nonwelded section were used (100 Btu/hr ft² °F and 1700 Btu/hr ft² °F). The lower limit of the "effective coefficient" of 100 Btu/hr ft² °F was calculated by assuming conduction and radiation through an effective steam gap 0.004 in. thick and 0.026 in. wide. The upper limit of 1700 Btu/hr ft² °F assumed contact over a 0.026-in. portion of the circumference. The actual condition was expected to be closer to the lower limit of the "effective coefficient."

The results of the more detailed calculation were correlated as a function of the channel heat transfer coefficient h and the "effective coefficient" between thermocouple and tube surface h gap as shown in Fig. 8. The correlation was obtained by assuming that heat transfer through the thermocouple is similar to heat transfer through a fin. For a rectangular fin, the ratio of temperature excess (above environment) of a point x in the fin to temperature excess of the fin base is (see Ref. 12):

$$\frac{\theta}{\theta_o} = \frac{\cosh \left[(x - L) \sqrt{h/ky_o} \right]}{\cosh (L \sqrt{h/ky_o})}$$

where L = length of the fin, ft

y_o = half-thickness of the fin, ft

k = thermal conductivity of the fin, Btu/hr ft °F

h = surface heat transfer coefficient, Btu/hr ft² °F

For a point $x = L/2$, as the argument of the cosh increases due to increasing h from 0 to large values, the ratio θ/θ_o approaches $1/\exp (L/2 \sqrt{h/ky_o})$ and the values are always greater than $1/\exp (L/2 \sqrt{h/ky_o})$. For $h = 0$, the ratio $\theta/\theta_o = 1$. The calculated points as shown in Fig. 8 fall close to exponential curves through the point $\theta_{t/c}/\theta_{tube} = 1.0$ for $h = 0$.

For radiation cooling, the average value of the heat transfer coefficient from the surface of the thermocouple is calculated to be 23 Btu/hr ft² °F for the inner fuel thermocouple and 29 Btu/hr ft² °F for the outer fuel thermocouple. The heat sink for each annulus is the outer surface of the annulus. Thus, for the inner fuel thermocouple (from Fig. 8):

$$\left(\frac{\theta_{tc}}{\theta_{tube}} \right)_I = \frac{(T_{tc})_I - T_O}{T_I - T_O} = 0.72 \text{ to } 0.92 \quad (7)$$

and for the outer fuel thermocouple,

$$\left(\frac{\theta_{tc}}{\theta_{tube}} \right)_O = \frac{(T_{tc})_O - T_S}{T_O - T_S} = 0.84 \text{ to } 0.97 \quad (8)$$

where subscripts tc refers to thermocouple; I, O, S refer to inner fuel tube, outer fuel tube and insulating tube, respectively. In the comparison of calculated and measured thermocouple temperatures, the lower value is used since it is conservative.

The poison tube thermocouple does not require a correction since heat in the poison pin is low and the thermocouple temperature is nearly the same as the inner fuel tube and poison tube. This fact is used in Sec. 6 to obtain an approximate experimental value of the correction factor for the inner fuel thermocouple to compare with calculated values given in Fig. 8.

6. EXPERIMENTAL RESULTS

Results are presented in Table 4 and Figs. 9 through 34. Table 4 gives measured temperatures, most probable calculated temperatures, and the ratio of linear heat rate deduced from measured temperatures to the most probable calculated linear heat rate. This ratio (Q_m^I/Q_C^I) is a measure of the accuracy of the heat transfer and flux distribution calculation models. Figures 9 through 31 compare measured and calculated temperature of each thermocouple as a function of measured power. Calculated temperatures for a factor of 0.75 and 1.33 on Q_C^I are also shown. Figures 32, 33, and 34 correlate measured temperatures with linear heat rate. The linear heat rate for measured temperatures is related to measured reactor power by the calculated flux distribution factors of Table 3.

6.1 COMPARISON OF CALCULATED AND EXPERIMENTAL RESULTS

The major purpose of the radiation cooling test was to determine the accuracy with which peak superheater fuel temperatures could be calculated with no steam flow through the superheater. One quantitative method of comparison is to compare the linear heat rate calculated from the measured temperature (Q_m^I) with the linear heat rate from physics calculations of flux distribution Q_C^I . The value Q_m^I is obtained from Fig. 7, using measured temperature. Figure 7 is calculated from the heat transfer equations of Sec. 5. The value of Q_C^I is obtained from Table 3, using measured power. Table 3 is the result

of physics calculations as described in Ref. 2. Both Fig. 7 and Table 3 are calculated using "most probable" or "best known" values. Thus, a ratio of $Q_m^i/Q_c^i = 1.0$ for a particular thermocouple indicates that both heat transfer and neutron flux distribution calculations are correct, and the calculated temperature equals the measured temperature. For values different from 1.0, the heat transfer calculations, flux distribution calculations, or a combination of the two are incorrect.

The calculated linear heat rate (Q_c^i) for Test 723 is known within a factor of 1.33. Reactor power is known within a factor of 1.10 based on power calibration tests (see Ref. 3). Peak heat generation, relative to the reactor power, is known within a factor of 1.21, based on flux mapping tests (see Ref. 13). It should be noted that the 1.21 factor applies to flux peaks and not local flux at any point in the core.

The accuracy with which Q_m^i is known depends on the temperature level and thermocouple location. For maximum temperatures in Test 723 (~700 F) on the poison tube and inner fuel tube which are of major significance, the accuracy of Q_m^i is estimated to be 1.10. Accuracy of thermocouples is ± 5 F for continuous thermocouples and ± 10 F for thermocouples with connectors. From Fig. 7 for 700 F on the poison tube, 5 F and 10 F correspond to a factor on Q_m^i of 1.03 and 1.05, respectively. Accuracy of the heat transfer model is estimated to be 1.05 for the poison tube thermocouple at 700 F. For the other thermocouples (inner fuel and outer fuel), the accuracy is not as good since they are in a larger radial temperature gradient compared to the poison tube thermocouple (see Fig. 7). The accuracy of Q_m^i for temperatures below 500 F is not as good. For the poison tube at 500 F, a difference of 10 F corresponds to a factor Q_m^i of 1.12 compared with 1.05 at 700 F.

For the following evaluations, the accuracy of the ratio Q_m^i/Q_c^i is taken as a factor of 1.33 (0.75 to 1.33), which is the accuracy of Q_c^i . This is the best estimate of the accuracy for the poison pin at the highest measured temperatures, which are of most significance. The first two runs in Table 4 are generally excluded from the evaluations, because the larger uncertainties at temperatures below 500 F may obscure real trends.

The expected and actual peak temperatures occur on the poison pin at locations 39 and 57 in., with inner fuel temperatures only slightly below poison pin temperatures. Examination of the ratio Q_m^i/Q_c^i in Table 4 for thermocouples P-57, I-57, P-39, and I-39 in the four hottest elements shows that peak temperatures are predicted very closely by the flux distribution and heat flow models described in Sec. 5. The average of runs 3 to 10 is tabulated below and the temperatures are plotted in the figures indicated.

<u>Fig. No.</u>	<u>Element</u>	<u>Thermocouple</u>	<u>(Q_m^i/Q_c^i) Avg.</u>
9	E-9	P-57	0.890
10	E-9	I-57	0.965
15	W-9	I-39	0.833
20	A-18	P-39	0.926
21	A-18	I-39	0.976
24	Z-1	I-39	0.874

Overall Average 0.911

For Test 723, the range of expected values of Q'_m/Q'_c including uncertainties was 0.75 to 1.33. The average of the test data near peak temperature locations is 9 percent less than the most probable value, and the maximum value (1.04) is considerably less than the product of the uncertainties in predicting Q'_c (1.33).

The measured temperatures at thermocouple locations which are not in the peak flux regions are also predicted within the expected accuracy, with some understandable exceptions. Those thermocouples in the lower 12 in. of the core were designed for steam flow data and not for radiation cooling data, but are included in Table 4 and data plots as part of the data obtained. These temperatures are expected to be lower than calculated possibly due to local heat convection at the ends of the element. Therefore, these thermocouples are not included in evaluation of the calculational model.

Considering the prediction of outer fuel tube thermocouple temperatures in the four hottest elements at 39 and 57 in., the measured values are slightly higher than predicted, but are still within expected accuracy. The ratio of Q'_m/Q'_c , averaged over runs 3 to 10 is tabulated below and a comparison of measured and calculated temperatures is plotted in the figures indicated.

<u>Fig. No.</u>	<u>Element</u>	<u>Thermocouple</u>	<u>$(Q'_m/Q'_c)'$ Avg.</u>
11	E-9	O-57	1.018
12	E-9	O-39	1.039
16	W-9	O-39	1.174
22	A-18	O-39	1.063

Overall Average 1.074

The maximum value of Q'_m/Q'_c is 1.23 and the minimum is 0.94. The range of values due to uncertainty in predicting Q'_c for Test 723 is 0.75 to 1.33. However, the average for the outer fuel tube (1.074) is higher than the average for the inner fuel and poison tubes (0.911). It is concluded that the heat transfer model underpredicts the outer fuel tube temperature. The flux levels for thermocouples at a given elevation are essentially the same (except for a small local factor for angular orientation) and, therefore, do not affect the comparison of outer fuel with inner fuel and poison tube thermocouples. The fact that outer fuel tube temperature is underpredicted is not significant in predicting peak temperatures, since outer fuel temperature is always less than inner fuel temperature during radiation cooling. At 0.97 Mwt (Run 9, Table 4) at 57 in. in element E-9, the poison, inner fuel, and outer fuel measured temperatures are 742 F, 725 F, and 659 F, respectively.

An evaluation of flux tilt in the core can be made by comparing the ratio of Q'_m/Q'_c for like thermocouples in symmetrical elements. The average values of Q'_m/Q'_c for O-39 and I-39 thermocouples are compared below.

<u>Element</u>	<u>Thermocouple</u>	<u>(Q_m^i/Q_c^i) Avg. (Table 4)</u>	<u>Core Quadrant (Figure 1)</u>
E-9	O-39	1.039	SE
W-9	O-39	1.174	NW
A-18	I-39	0.976	SE
Z-1	I-39	0.874	NW

The above data indicate no appreciable flux tilt. While the outer fuel thermocouples (O-39) show a higher measured value in the NW quadrant, the inner fuel values show a higher measured value in the SE quadrant. The average of the values in the SE quadrant is 1.008 compared with 1.024 in the NW quadrant. The measured temperatures for elements W-9 and Z-1 are expected to be less accurate than the other two because these elements have connections in the reactor.

A comparison of the thermocouple temperatures at 21 in. in the hot element (E-9) shows that the measured values are considerably less than calculated values.

<u>Fig. No.</u>	<u>Element</u>	<u>Thermocouple</u>	<u>(Q_m^i/Q_c^i) Avg.</u>
13	E-9	P-21	0.681
14	E-9	I-21	0.640

Overall Average 0.661

This is believed to result from a lower neutron flux at this location than calculated. This is consistent with flux mapping tests (see Ref. 13) which indicated that while axial flux peaks in the superheater are predicted fairly accurately, the measured flux in the lower part of the core is less than calculated for many of the runs.

The measured temperatures in element K-10, near the core center, are all significantly less than calculated. Average values of Q_m^i/Q_c^i for runs 3 to 10 are given below, except for P-21. This thermocouple read approximately reactor water temperature for most of the runs and varied for others (see Table 4).

<u>Fig. No.</u>	<u>Element</u>	<u>Thermocouple</u>	<u>(Q_m^i/Q_c^i) Avg.</u>
27	K-10	I-39	0.293
28	K-10	O-39	0.488
30	K-10	I-21	0.235
31	K-10	O-21	0.541

Overall Average 0.389

The lower temperatures in element K-10 are believed to result from lower neutron flux in this element than calculated. Operation with steam flow has shown that the measured power fraction is less than calculated. For a small difference in power fraction, the difference in power of the central elements is large, since power in the elements near the

boiler is not affected appreciably by differences in power fraction. Another possible explanation of lower temperatures in the central element is steam flow through the superheater, either by recirculation or net steam flow downward due to steam leakage past isolation valves. In either case, flow through the central element would be expected to be downward and the ratio (Q'_m/Q'_c) would be expected to be larger for thermocouples at 21 in. compared with those at 39 in. Comparison of values for I-39 with I-21 and O-39 with O-21 does not show any significant difference. It is concluded that heat transport by steam flowing through the elements is negligible compared with heat transfer by radiation and conduction to the moderator; and that the lower than expected temperatures in K-10 are due to lower neutron flux.

A comparison of all measured temperatures at 21, 39, and 57 in. with the most probable calculated temperatures is given in Figs. 32, 33, and 34. The linear heat rate for measured temperatures is related to measured power by the calculated flux distribution factors of Table 3. From Fig. 32 for the poison tube, it is seen that the measured peak temperature is always less than the calculated value. For the inner fuel tube (Fig. 33), a few of the measured temperatures are slightly higher than calculated (6 F), but the calculational model predicts inner fuel temperatures closely. For the outer fuel tube (Fig. 34), the model underpredicts the thermocouple temperature, as noted previously. However, the most significant prediction is the peak temperature which occurs on the poison tube (Fig. 32).

Figures 32, 33, and 34 also show that the variation of superheater temperature with power is adequately predicted by the calculational model. The data follows the shape of the calculated curve, which is slightly concave downward on these plots. Therefore, calculation of temperatures at higher power levels will be sufficiently accurate to predict peak superheater temperatures during reactor shutdown up to 1300 F.

6.2 EXPERIMENTAL CHECK OF THERMOCOUPLE TEMPERATURE CORRECTION FACTOR

A check on the accuracy of the calculated thermocouple correction factor for the inner fuel thermocouple is obtained by comparing the poison tube thermocouple temperature with the inner fuel tube thermocouple at the same elevation in an element. The poison tube thermocouple reading is assumed to be the inner fuel tube temperature, since the radial temperature gradient in the annulus between the inner fuel and poison tube is small (see Fig. 7). Poison tube and inner fuel tube thermocouple temperatures are obtained from the data plotted in Figs. 32 and 33 at a linear heat rate of 300 Btu/(hr)(ft). The temperature difference between inner and outer fuel tubes is the calculated temperature difference from Fig. 7 at the inner fuel tube temperature. From equation (7), the calculated value of inner fuel thermocouple temperature is:

$$\left(\frac{\theta_{tc}}{\theta_{tube}} \right)_i = \frac{T_{tc} - T_o}{T_i - T_o} = 0.72 \text{ to } 0.92$$

The values computed from measured temperatures are tabulated below.

Element	Distance Above Bottom of Fuel Element, in.	Temperatures, °F				$\frac{\theta_{tc}}{\theta_{tube}}$
		T_{tc} (Fig. 33)	T_i (Fig. 32)	$T_i - T_o$ (Fig. 7)	T_o	
A-18	39	662	674	62	612	0.81
E-9	21	592	604	47	557	0.75
E-9	57	660	664	64	600	0.94

It is seen that values calculated from measured data compare well with the calculated range of values for

$$\frac{\theta_{t/c}}{\theta_{tube}}.$$

The value used in the analysis was 0.72, which results in the lowest expected thermocouple temperature.

7. METHOD FOR CALCULATING TEMPERATURES AFTER REACTOR SHUTDOWN

The method for calculating superheater temperature following a reactor shutdown is similar to the method used in Test 723. In this case, the reactor power after shutdown is calculated instead of measured. The flux distribution in the reactor after shutdown is determined from the distribution which existed during the operating period immediately preceding the shutdown. Also, the distribution of heating among the components of the fuel element are different for fission product heating compared to fission heating. The data required for calculating temperatures following a reactor shutdown are given in the following sections.

The curves given here are specifically for the initial core loading of 411 fuel elements and initial control rod programs. However, they may be extended to other loadings and control rod programs with appropriate corrections.

7.1 FISSION PRODUCT DECAY HEAT

Fission product decay heat (P_s) as a function of operating power level (P_o), time at power (T_o), and time after shutdown (T_s) is given in Fig. 35 (see Ref. 14). The estimated accuracy is ± 10 percent for shutdown times greater than 100 sec. The fission product decay heat P_s is that portion of the total β and γ heat which is transferred through the steam annuli. Of the total decay heat, all of the β and 78 percent of the γ heat is released inside the process tube.

The operating power level may not be constant over the significant portion of time preceding reactor shutdown. To calculate decay power for a variable power history, the curves of Fig. 35 should be applied for each time period over which the power is substantially constant. The decay power is obtained for each period of constant power operation and then added to obtain total decay power.

The distribution of heating in the core is proportional to the distribution during power operation. Flux peaks for various control rod positions were calculated for initial startup tests (see Refs. 15, 16). The peak linear heat rate in the superheater during power operation is plotted in Fig. 36 for the range of control rod positions during power operation. Steam voids in the boiler which are a function of power level have some influence on flux peaks but not as much as control rod position. The values in Fig. 36 are the highest for the possible range of power levels at a given rod position.

The linear heat rate in Fig. 36 is for 1 Mwt of operating reactor power. To obtain shutdown linear heat rate in the superheater, Q'_s , multiply the value from Fig. 36 times the operating reactor power level times the ratio P_s/P_o from Fig. 35.

7.2 SUPERHEATER TEMPERATURES

The heat transfer model is the same as described in Sec. 5.1 except that the distribution of heating between superheater element components is different for fission product heating from the distribution for fission heating. The distribution of heat flow across the four steam annuli is:

$$q'_p = 0.120 Q'_s \quad (9)$$

$$q'_i = 0.498 Q'_s \quad (10)$$

$$q'_o = 0.940 Q'_s \quad (11)$$

$$q'_s = 1.000 Q'_s \quad (12)$$

where Q'_s is the linear heat rate during shutdown, Btu/(hr)(ft) calculated as in Sec. 7.1. Curves for superheater temperature as a function of linear heat rate are given in Fig. 37 for moderator water temperatures of 400 and 500 F. For water temperatures between these values, linear interpolation may be used. Curves are limited to 1300 F which is the maximum design fuel temperature.

7.3 ACCURACY OF CALCULATION

The curves given in Figs. 35, 36, and 37 are based on most probable values. To obtain an upper limit on peak fuel temperatures, the following factors should be applied to Q'_s .

<u>Uncertainty</u>	<u>Factor on Q'_s</u>
1. Decay power level	1.10
2. Flux distribution	1.21
3. U-235 concentration	<u>1.15</u>
Product	1.54

Item 3 is included for prediction of maximum temperatures after shutdown; whereas, it was not included in the prediction of maximum temperatures for the test. The probability that high U-235 concentration will exist in a peak flux region is greater than the probability that it existed at one of the thermocouple locations. The uncertainty margin of 1.54 on Q'_s corresponds to a temperature difference of 190 F for the design temperature limit of 1300 F on the poison pin (see Fig. 37).

The maximum value of Q'_m/Q'_c based on measured temperatures during Test 723 was 0.99 for the poison tube, 1.04 for an inner fuel tube, and 1.23 for an outer fuel tube. While the maximum ratios of measured to calculated heat rates, particularly for the poison and inner fuel tube, are considerably less than the recommended uncertainty factor, the larger factor is believed necessary for predicting an upper limit for shutdown cooling. During the test, only one control rod configuration was tested, for which specific flux calculations were available. For shutdown cooling, interpolated flux peaking factors are used, and decay heat peaking factors may be influenced by several rod positions during the preceding operating period.

8. REFERENCES

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9. TABLES

1. Results of Test 711A, Thermocouple Calibration Test
2. Superheater Thermocouple Calibration Checks
3. Fuel Element Linear Heat Rate at Thermocouple Locations During Test 723
4. Measured and Calculated Temperatures During Radiation Cooling Test 723

TABLE 1

RESULTS OF TEST 711A, THERMOCOUPLE CALIBRATION TEST

Reactor Condition		Equilibrium - Superheater Flooded								Cooling Down Superheater Drained	
Reactor Water Temperature, °F		150	200	245	303	346	400	426	478	~450	~355
Core Location	Thermocouple Designation	Superheater Thermocouple Temperature, °F									
E-9	O-21	82	86	86	88	86	99	86	96	449	351
	P-21	150	197	242	305	345	400	423	476	454/447	359/353
	I-21	151	199	243	306	347	401	426	478	453	355
	P-57	151	199	243	306	346	401	425	478	454	357
	O-57	152	198	243	306	347	401	426	479	451	352
	I-57	151	199	243	307	347	401	426	478	451	355
	P-39	0	0	0	0	0	0	0	0	0	0
	O-39	150	198	243	305	346	401	425	478	449	351
	I-39	0	0	0	0	0	0	0	0	0	0
K-10	O-21	153	264	0	varied	586	541	varied	542	515	422
	P-21	130	189	varied	0	444	435	varied	442	484	382
	I-21	156	200	251	337	389	427	varied	501	452	360
	P-57	0	0	0	0	0	0	0	0	0	0
	O-57	157	216	varied	313	412	varied	varied	480	485	439
	I-57	150	197	0	varied	varied	varied	full scale	varied	806	589
	P-39	full scale	full scale	full scale	full scale	full scale	full scale	full scale	full scale	708	502
	O-39	169	229	varied	varied	varied	varied	0	502	504	395
	I-39	161	208	260	332	varied	varied	varied	480	450	364
Z-1	I-00	0	0	0	0	0	full scale	full scale	varied	195	0
	I-04	154	204	274	337	varied	varied	varied	507	471	364
	O-06	0	0	0	0	0	0	0	0	0	0
	P-08	161	221	279	344	varied	varied	varied	498	448	350
	O-10	153	226	305	varied	varied	0	0	443	451	359
	P-10	162	211	255	479	varied	557	425	476	456	369
	P-39	0	0	0	0	0	0	0	0	0	0
	O-39	0	0	0	0	0	0	0	0	0	0
	I-39	151	196	253	311	varied	varied	varied	457	456	365
W-9	I-00	156	212	283	324	varied	varied	varied	498	459	372
	I-04	158	214	varied	340	357	varied	varied	484	458	364
	O-06	157	277	varied	353	varied	varied	varied	502	488	381
	P-08	0	0	0	0	0	0	0	0	0	0
	O-10	163	229	283	338	varied	varied	varied	404	433	359
	P-10	0	0	0	0	0	0	0	0	0	0
	P-39	152	247	varied	varied	353	409	varied	482	449	359
	O-39	147	198	varied	varied	varied	489	varied	498	459	380
	I-39	147	205	varied	532	440	451	varied	453	451	362
A-18	I-00	148	199	243	306	346	401	424	474	451/443	359/354
	I-04	0	0	0	0	0	0	0	0	0	0
	O-06	148	199	243	306	345	400	424	475	458	363
	P-08	156	200	292	323	345	400	varied	480	455	366
	O-10	148	199	243	306	345	400	424	473	448	353
	P-10	151	199	234	varied	346	400	varied	475	454	366
	P-39	150	199	244	306	346	400	425	475	454	359
	O-39	150	200	244	306	346	400	425	475	449	354
	I-39	150	200	244	306	346	401	426	475	451	357

TABLE 2

SUPERHEATER THERMOCOUPLE CALIBRATION CHECKS

(Superheater thermocouple temperatures which differ from reactor water temperature by more than 5 F are noted.)

Date	Feb. 1966	Feb. 1966	Feb. 1966	Mar. 7, 1966	Mar. 7, 1966	14:00 Mar. 21, 1966	08:30 Mar. 22, 1966	09:00 Mar. 23, 1966	09:30 Mar. 24, 1966	Rating
Test	Element Immersed in Hot Water	Reactor Leads Checked	Recorder Checked Equilibrium (Drained)	Test 711A Equilibrium (Flooded)	Test 251 Cooldown (Drained)	Test 723 Prior to Startup (Drained)	Test 723 Prior to Startup (Drained)	Test 723 Prior to Startup (Drained)	Test 723 Prior to Startup (Drained)	at End of Test 723
Reactor Water Temperature, T_R , °F			ambient	150, 200, 245, 303, 346, 400, 426, 480	~355 and ~450	417	422	412	422	
Core Location	Thermocouple Designation			(See Table 1)	(See Table 1)	(Note 1)	(Note 1)	(Note 1)	(Note 3)	(Note 4)
E-9	O-21 P-21 I-21 P-57 O-57 I-57 P-39 O-39 I-39	ambient ambient	<ambient	<< T_R		<< T_R (Note 2)	<< T_R (Note 2)	<< T_R	<< T_R	failed good good ? good failed good failed
K-10	O-21 P-21 I-21 P-57 O-57 I-57 P-39 O-39 I-39	ambient shorted open	zero	varied varied varied zero varied varied >ambient varied varied	+65 +35 zero varied varied full scale varied varied	+22 +19 +15 zero zero varied >> T_R +23 +18	+35 +35 +27 zero +19 varied varied +35 +27	+45 +38 +30 zero +21 varied varied +30 +26	-9 zero varied varied zero	fair fair poor failed failed failed failed fair fair
Z-1	I-00 I-04 O-06 P-08 O-10 P-10 P-39 O-39 I-39	open open open open open	zero	varied varied zero varied varied zero zero zero varied	varied +15 zero +10 zero zero zero +10	full scale (Note 2) full scale +18 >> T_R zero zero << T_R	full scale (Note 2) varied +33 +21 >> T_R zero zero +23	full scale +36 varied +30 +18 >> T_R zero zero +23	zero -9 varied -13 >> T_R zero zero zero	failed poor failed fair failed failed failed failed fair
W-9	I-00 I-04 O-06 P-08 O-10 P-10 P-39 O-39 I-39	 open open	zero	varied varied varied zero varied zero varied varied varied	+15 +8 +35 zero -20 zero +20	 +8 +11 zero zero zero zero +18	+23 +27 +22 zero +31 zero +18 +32 +18	+23 +23 +23 zero +29 zero +18 +31 +18	zero zero +13 zero zero +13	fair fair fair failed poor failed failed poor fair
A-18	I-00 I-04 O-06 P-08 O-10 P-10 P-39 O-39 I-39		<ambient	zero	zero	zero	zero	zero	zero	good failed failed good good failed good good good

NOTES:

- (1) Ice junctions for multipoint recorders outside of insulated container in ice bath.
- (2) These thermocouples were connected to control room recorders which had ice junction inside of insulated container. Readings were within 5 F of water temperature.
- (3) All ice junctions in insulated container.
- (4) Good - consistently within 5 F; continuous leads.
Fair - within 5 F after correcting ice junction error; connectors.
Poor - within 15 F after correcting ice junction error; connectors.
Failed - zero, full scale or varying.

TABLE 3

FUEL ELEMENT LINEAR HEAT RATE AT THERMOCOUPLE LOCATIONS DURING TEST 723

- Reactor Conditions: 1. Superheater power fraction = 0.1589
 2. Zero steam voids in boiler
 3. CRG I and II at 73 in., III at 27 in., IV and V at 0 in.

Core Location	Thermocouple Designation	Radial Flux Ratio ϕ_R	Axial Flux Ratio ϕ_A	Local Flux Ratio ϕ_L	Linear Heat Rate, Btu/hr ft Per Mwt Reactor Power Q'/Q_R
E-9	P-57	1.61	1.18	1.04	435
	I-57	1.61	1.18	1.00	418
	O-57	1.61	1.18	1.04	435
	O-39	1.61	1.24	0.96	422
	P-21	1.61	1.04	1.00	368
	I-21	1.61	1.04	1.05	387
W-9	I-39	1.61	1.24	1.00	440
	O-39	1.61	1.24	0.97	427
	O-10	1.61	0.80	1.06	300
	O-06	1.61	0.60	1.00	212
	I-04	1.61	0.51	1.05	189
	I-00	(outlet steam thermocouple)			
A-18	P-39	1.51	1.14	0.96	364
	I-39	1.51	1.14	0.97	368
	O-39	1.51	1.14	0.96	364
	O-10	1.51	0.93	1.00	309
	I-00	(outlet steam thermocouple)			
Z-1	I-39	1.51	1.14	1.00	379
	P-08	1.51	0.82	0.94	256
	I-04	1.51	0.59	1.00	196
K-10	I-39	0.774	1.19	1.00	202
	O-39	0.774	1.19	1.00	202
	P-21	0.774	1.19	1.00	202
	I-21	0.774	1.19	1.00	202
	O-21	0.774	1.19	1.00	202

TABLE 4

MEASURED AND CALCULATED TEMPERATURES DURING RADIATION COOLING TEST 723

Run No.	1	2	3	4	5	6	7	8	9	10																					
Date	Mar. 22, 1966						Mar. 23, 1966																								
Time	0945	1105	1330	1410	1510	1630	1305	1430	1530	1945																					
Reactor Power, Mwt	0.12	0.18	0.32	0.40	0.49	0.64	0.79	0.97	0.97	0.46																					
Reactor Water Temp., °F	424	424	424	424	424	424	422	422	421	422																					
Control Rod Group III, in.	25.7	25.8	25.9	25.9	26.0	26.0	25.5	25.6	-	25.8																					
Calculated Peak Temp., °F																															
(a) Most Probable Q'	484	514	575	605	637	686	731	781	781	627																					
(b) 1.33 Factor on Q'	504	540	615	660	691	749	804	860	860	680																					
Core Location	Thermocouple Designation	T _m	T _c	$\frac{Q'_m}{Q'_c}$	T _m	T _c	$\frac{Q'_m}{Q'_c}$	T _m	T _c	$\frac{Q'_m}{Q'_c}$	T _m	T _c	$\frac{Q'_m}{Q'_c}$	T _m	T _c	$\frac{Q'_m}{Q'_c}$	T _m	T _c	$\frac{Q'_m}{Q'_c}$	T _m	T _c	$\frac{Q'_m}{Q'_c}$	T _m	T _c	$\frac{Q'_m}{Q'_c}$	T _m	T _c	$\frac{Q'_m}{Q'_c}$	T _m	T _c	$\frac{Q'_m}{Q'_c}$
E-9	P-57	460	484	0.67	488	510	0.78	539	565	0.80	560	692	0.78	610	622	0.92	650	667	0.91	695	710	0.92	740	757	0.93	742	757	0.94	599	612	0.92
	I-57	455*	468	0.70	478*	493	0.74	530*	544	0.86	548*	570	0.82	596*	597	0.99	634*	638	0.97	685	678	1.04	726	722	1.02	725	722	1.02	589	589	1.00
	O-57	454	457	0.94	471	474	0.93	510	511	0.98	526	531	0.94	564	551	1.11	591	584	1.05	598	615	0.89	660	650	1.05	659	650	1.05	552	545	1.07
	O-39	456	455	0.99	474	473	1.05	507	509	0.97	523	528	0.95	558	548	1.08	583	580	1.02	624	611	1.08	662	645	1.09	662	645	1.09	546	542	1.03
	P-21	458	475	0.86	465	498	0.68	493	547	0.61	510	571	0.62	540	598	0.70	565	637	0.68	597	676	0.70	635	718	0.73	638	718	0.74	525	555	0.67
	I-21	450	463	0.65	466	488	0.66	494	536	0.59	508	560	0.58	538	586	0.66	563	625	0.64	594	664	0.65	629	705	0.67	633	705	0.69	525	577	0.64
W-9	I-39	457*	472	0.72	478*	497	0.71	523*	548	0.76	542*	575	0.74	586*	604	0.87	619*	647	0.84	660*	689	0.86	700*	733	0.86	710	733	0.90	570	595	0.83
	O-39	(480)*	456	1.72	482*	473	1.17	523*	510	1.17	537*	530	1.08	574*	550	1.23	606*	582	1.18	638*	613	1.17	675*	647	1.15	682	647	1.18	567	544	1.23
	O-10	451*	448	1.01	458*	461	1.01	486*	486	0.89	492*	500	0.90	507*	516	0.90	523*	540	0.83	537*	565	0.77	549*	591	0.71	-	591	-	488	511	0.72
	O-06	435*	440	0.86	438*	450	0.68	445*	470	0.53	453*	481	0.56	462*	492	0.57	467*	509	0.49	468*	528	0.41	479*	550	0.37	487	550	0.47	451	488	0.44
	I-04	440*	446	0.87	442*	457	0.64	448*	482	0.44	452*	495	0.41	457*	509	0.38	462*	533	0.34	464*	557	0.30	472*	582	0.27	-	582	-	451	505	0.34
	I-00	424*	-	-	423*	-	-	422*	-	-	423*	-	-	422*	-	-	422*	-	-	420*	-	-	420*	-	-	426	-	-	424	-	-
A-18	P-39	464	477	0.87	484	500	0.87	526	546	0.84	547	570	0.83	586	596	0.94	623	637	0.94	665	675	0.95	715	716	0.99	709	716	0.97	580	587	0.96
	I-39	458	463	0.84	480	485	0.88	523	531	0.94	542	554	0.89	581	579	1.01	616	617	0.91	658	653	1.02	698	694	1.03	698	694	1.03	569	571	0.98
	O-39	453	450	1.07	466	466	1.00	498	498	1.01	511	515	0.96	543	533	1.10	570	562	1.07	602	590	1.09	637	620	1.11	636	620	1.11	533	527	1.05
	O-10	442	447	0.87	448	460	0.72	464	487	0.64	471	502	0.59	486	518	0.64	500	543	0.61	518	568	0.63	538	595	0.62	540	595	0.64	478	513	0.60
	I-00	426	-	-	424	-	-	426	-	-	426	-	-	424	-	-	424	-	-	422	-	-	422	-	-	421	-	-	419	-	-
Z-1	I-39	461*	467	0.84	480*	490	0.81	519*	539	0.80	537*	564	0.78	579*	590	0.92	613*	630	0.90	646*	669	0.88	688*	711	0.90	693	711	0.92	567	582	0.89
	P-08	450*	459	0.85	461*	477	0.78	481*	513	0.67	491*	532	0.63	508*	553	0.64	525*	583	0.60	540*	613	0.56	553*	645	0.52	571	645	0.60	501	546	0.63
	I-04	445	448	1.06	447	460	0.76	449*	475	0.48	453*	497	0.41	462*	512	0.44	467*	536	0.37	485*	560	0.41	500*	586	0.42	493	586	0.38	462	507	0.47
K-10	I-39	424*	446	-	428*	459	0.22	436*	486	0.23	437*	501	0.21	449*	517	0.30	452*	541	0.26	466*	566	0.28	482*	592	0.31	493	592	0.37	453	511	0.38
	O-39	429*	439	-	434*	447	0.60	441*	466	0.45	442*	476	0.40	454*	487	0.49	454*	504	0.38	468*	522	0.44	483*	543	0.49	493	543	0.61	455	484	0.54
	P-21	421*	453	-	425*	467	0.11	428*	497	0.09	426*	512	0.05	429*	529	0.06	429*	555	0.05	(442)*	581	0.11	470*	609	0.22	(536)	609	0.56	438	524	0.17
	I-21	420*	446	-	424*	459	-	432*	486	0.17	433*	501	0.15	445*	517	0.25	445*	541	0.23	465*	566	0.28	487*	592	0.33	-	592	-	-	511	-
	O-21	428*	439	-	434*	447	0.58	441*	466	0.46	445*	476	0.45	455*	487	0.50	461*	504	0.45	477*	522	0.52	492*	543	0.59	519	543	0.79	457	484	0.57

NOTES:

T_m = measured temperature, °F; starred values are corrected for ice junction drift; values in parentheses showed slight variation during run.

T_c = calculated temperature, °F, using most probable value of Q' (Table 3).

Q'_m = linear heat rate from Fig. 7 using measured temperature, Btu/hr ft length of fuel element.

Q'_c = calculated most probable value of linear heat rate from Table 3, Btu/hr ft.

10. FIGURES

1. Location of Instrumented Fuel Elements in Superheater Core
2. Location of Thermocouples for Radiation Cooling (Type R)
3. Location of Thermocouples for Steam Cooling (Type S)
4. Thermocouple Spot Welded to Poison Tube, Transverse
5. Thermocouple Welded to Large Diameter Fuel Tube
6. Thermocouple Welded to Small Diameter Fuel Tube
7. Superheater Fuel Temperature vs. Linear Heat Rate
8. Thermocouple Temperature
9. Measured Temperature vs. Power (fuel element E-9 and thermocouple P-57)
10. Measured Temperature vs. Power (fuel element E-9 and thermocouple I-57)
11. Measured Temperature vs. Power (fuel element E-9 and thermocouple O-57)
12. Measured Temperature vs. Power (fuel element E-9 and thermocouple O-39)
13. Measured Temperature vs. Power (fuel element E-9 and thermocouple P-21)
14. Measured Temperature vs. Power (fuel element E-9 and thermocouple I-21)
15. Measured Temperature vs. Power (fuel element W-9 and thermocouple I-39)
16. Measured Temperature vs. Power (fuel element W-9 and thermocouple O-39)
17. Measured Temperature vs. Power (fuel element W-9 and thermocouple O-10)
18. Measured Temperature vs. Power (fuel element W-9 and thermocouple O-06)
19. Measured Temperature vs. Power (fuel element W-9 and thermocouple I-04)
20. Measured Temperature vs. Power (fuel element A-18 and thermocouple P-39)
21. Measured Temperature vs. Power (fuel element A-18 and thermocouple I-39)
22. Measured Temperature vs. Power (fuel element A-18 and thermocouple O-39)
23. Measured Temperature vs. Power (fuel element A-18 and thermocouple O-10)
24. Measured Temperature vs. Power (fuel element Z-1 and thermocouple I-39)
25. Measured Temperature vs. Power (fuel element Z-1 and thermocouple P-08)
26. Measured Temperature vs. Power (fuel element Z-1 and thermocouple I-04)
27. Measured Temperature vs. Power (fuel element K-10 and thermocouple I-39)
28. Measured Temperature vs. Power (fuel element K-10 and thermocouple O-39)
29. Measured Temperature vs. Power (fuel element K-10 and thermocouple P-21)
30. Measured Temperature vs. Power (fuel element K-10 and thermocouple I-21)
31. Measured Temperature vs. Power (fuel element K-10 and thermocouple O-21)
32. Poison Tube Temperatures
33. Inner Fuel Tube Temperatures
34. Outer Fuel Tube Temperature
35. Fission Product Decay Heat in Superheater
36. Maximum Linear Heat Rate vs. Control Rod Position
37. Linear Heat Rate vs. Poison Tube Temperature
- 38a. Thermocouple Leads in Reactor
- 38b. Thermocouple Leads in Reactor
39. End Fittings of Instrumented Elements
40. Thermocouple Leads Above Superheater Core

Roman numerals designate control rod group number.

Superheater control rods (Group I) are indicated by solid circles.

A-18, E-9, K-10, Z-1, W-9 are core locations which contain instrumented fuel elements.

Type S has thermocouples located for steam cooling.

Type R has thermocouples located for radiation cooling.

LOCATION OF INSTRUMENTED FUEL ELEMENTS IN SUPERHEATER CORE

FIG. 1

In thermocouple designation, letter indicates tube to which thermocouple is welded and number indicates axial location of hot junction in inches above bottom of poison rod end cap.

direction of T/C O-39, degrees
clockwise from north

core location	degrees
E-9	340
K-10	210

LOCATION OF THERMOCOUPLES FOR RADIATION COOLING (TYPE R)

FIG. 2

In thermocouple designation, letter indicates tube to which thermocouple is welded and number indicates axial location of hot junction in inches above bottom of poison end cap.

The hot junction of I-00 is 1.063 in. below the poison end cap and measures temperature of steam from all three coolant annuli.

direction of T/C O-39, degrees clockwise from north

core location	degrees
A-18	340
Z-1	30
W-9	90

LOCATION OF THERMOCOUPLES FOR STEAM COOLING (TYPE S)

FIG. 3

NL-15483
M-11435

Etchant:
Vilella's

Magn. 100X
20 Oct. 1964

THERMOCOUPLE SPOT WELDED TO POISON TUBE, TRANSVERSE

FIG. 4

NL-15770
M-11433

Etchant:
Vilella's

Magn. 100X
20 Oct. 1964

THERMOCOUPLE WELDED TO LARGE DIAMETER FUEL TUBE

FIG. 5

NL-15771
M-11434

Etchant:
Vilella's

Magn. 100X
20 Oct. 1964

THERMOCOUPLE WELDED TO SMALL DIAMETER FUEL TUBE

FIG. 6

SUPERHEATER FUEL TEMPERATURE VS. LINEAR HEAT RATE

FIG. 7

θ_{TC} = Temperature of thermocouple hot junction above heat sink.

θ_{tube} = Temperature of fuel tube above heat sink.

h = Fuel channel heat transfer coefficient.

h_{gap} = Effective coefficient between thermocouple and fuel tube over non-welded portion.

Points ($\circ$, Δ) are from analog calculations.

THERMOCOUPLE TEMPERATURE

FIG. 8

MEASURED TEMPERATURE VS. POWER

FIG. 9

MEASURED TEMPERATURE VS. POWER

FIG. 10

MEASURED TEMPERATURE VS. POWER

FIG. 11

MEASURED TEMPERATURE VS. POWER

FIG. 12

MEASURED TEMPERATURE VS. POWER

FIG. 13

MEASURED TEMPERATURE VS. POWER

FIG. 14

MEASURED TEMPERATURE VS. POWER

FIG. 15

MEASURED TEMPERATURE VS. POWER

FIG. 16

MEASURED TEMPERATURE VS. POWER

FIG. 17

MEASURED TEMPERATURE VS. POWER

FIG. 18

MEASURED TEMPERATURE VS. POWER

FIG. 19

MEASURED TEMPERATURE VS. POWER

FIG. 20

MEASURED TEMPERATURE VS. POWER

FIG. 21

MEASURED TEMPERATURE VS. POWER

FIG. 22

MEASURED TEMPERATURE VS. POWER

FIG. 23

MEASURED TEMPERATURE VS. POWER

FIG. 24

MEASURED TEMPERATURE VS. POWER

FIG. 25

MEASURED TEMPERATURE VS. POWER

FIG. 26

MEASURED TEMPERATURE VS. POWER

FIG. 27

MEASURED TEMPERATURE VS. POWER

FIG. 28

MEASURED TEMPERATURE VS. POWER

FIG. 29

MEASURED TEMPERATURE VS. POWER

FIG. 30

MEASURED TEMPERATURE VS. POWER

FIG. 31

POISON TUBE TEMPERATURES

FIG. 32

INNER FUEL TUBE TEMPERATURES

FIG. 33

FISSION PRODUCT DECAY HEAT IN SUPERHEATER

FIG. 35

Note: During control rod interchange, maximum linear heat rate in superheater is 505 Btu/hr-ft/Mwt.

MAXIMUM LINEAR HEAT RATE VS. CONTROL ROD POSITION

FIG. 36

LINEAR HEAT RATE VS. POISON TUBE TEMPERATURE

FIG. 37

THERMOCOUPLE LEADS IN REACTOR

FIG. 38a

THERMOCOUPLE LEADS IN REACTOR

FIG. 38b

END FITTINGS OF INSTRUMENTED ELEMENTS

FIG. 39

THERMOCOUPLE LEADS ABOVE SUPERHEATER CORE

FIG. 40