PIQUA NUCLEAR POWER FACILITY
SURVEILLANCE AND RECOVERY PROGRAM
FINAL REPORT

Editor
R. R. EGGLESTON

LEGAL NOTICE

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

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

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

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

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN ROCKWELL CORPORATION

CONTRACT: AT(04-3)-701
ISSUED: JUNE 15, 1968

DISTRIBUTION

This report has been distributed according to the category "General, Miscellaneous, and Progress Reports," as given in the Standard Distribution for Unclassified Scientific and Technical Reports, TID-4500.

DISCLAIMER

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

DISCLAIMER

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

CONTENTS

	Page
I. Introduction.	5
A. Plant Description.	5
B. Operation History	5
C. Piqua Shutdown	14
II. Recovery Program.	15
A. Core and Coolant Cleanup	16
B. Core and Coolant System Modifications	20
C. Control Rod Modifications	33
III. Coolant Technology.	41
A. Oxidation Experiments	41
B. Thermal Decomposition Experiments	43
IV. Surveillance Program.	45
V. Status at Program Termination.	49
A. Plant Status	49
B. Fuel Element Revision	49
C. Control Rod Drives	50
Reverences	53

TABLES

1. PNPF Reactor Operations Summary	7
2. Reactor Core Characteristics Comparison	28
3. Clean-Core Clean-Element Thermal Operation at 30 Mw.	31
4. Control-Rod Drive-Power Coil Modifications	32
5. Summary of Control Rod Drive Test Program.	35
6. Surveillance Program.	46

FIGURES

	Page
1. Reactor Core Vessel.	6
2. Operations Summary.	8
3. Core Loading Map	10
4. Quarter Section Test, 41-Modular Element Configuration	21
5. Fuel Element Lower End Fitting with Pressure Probes.	22
6. Seven-Element Test Assembly	23
7. Modified Fuel Element Lower End.	24
8. Reactor Elevation, 41-Element Restart Core	26
9. Core I and Restart Core Fuel Elements	27
10. Film Thickness at Hot Spot	30
11. Modified Control Rod	34
12. Status of Modified CRD Parts at Program Termination.	51

I. INTRODUCTION

The Piqua Nuclear Power Facility (PNPF) is an Atomic Energy Commission owned plant built as part of the AEC Power Demonstration Program. The plant was operated by the City of Piqua. Atomics International was the prime contractor during the design, construction, and startup of the plant. This report briefly describes the plant, its operational history to the time that the coke formation in the core was detected, the subsequent engineering evaluation, and the recovery program. Also included are brief descriptions of the coolant technology work and the Surveillance Program which were performed in support of the recovery program. Finally, the status of the plant and the Recovery Program effort at the time of the PNPF Program cancellation is summarized.

A. PLANT DESCRIPTION

The Piqua Nuclear Power Facility (PNPF) was a 45.5 Mwt reactor system for providing superheated steam at 550°F, 450 psia and at a rate of 150,000 lb/hr to the Piqua Municipal Power Station. The reactor, see Figure 1, was fueled with slightly enriched uranium alloy fuel for a total U-235 inventory of 94.4 kg. The reactor coolant and moderator was an organic hydrocarbon liquid consisting of a mixture of ortho-, meta-, and para-terphenyls (OMP). The flow rate of the coolant through the reactor vessel was 14,000 gpm, the coolant outlet and inlet temperatures were 570 and 525 °F respectively. A more complete description of reactor and facility together with a site description is available in the PNPF Final Safeguard Summary Report.⁽¹⁾

B. OPERATION HISTORY

The electrical power generation over the operating life of the plant is presented diagrammatically in Figure 2. Causes for the more extensive shutdowns during the operational life are also noted. The thermal energy generation is summarized in Table 1. The semiannual Reactor Operations Analysis Reports are referenced as appropriate and should be used for obtaining detailed operating history.

7-7710-043-14A

Figure 1. Reactor Core Vessel

TABLE 1
PNPF OPERATIONS SUMMARY
(thermal energy release)

Year Month	Thermal Energy Release (Mwh)	Accumulated Energy Release (Mwh)	Thermal Power (Mwt, avg/peak)		Avail- ability for Period (%)	Accumulated Availability (from 1/27/64) (%)
1964						
Jan 27	3,030	13,288	25.3	46.2	100.0	100.0
Feb	18,528	31,816	26.6	47.4	100.0	100.0
Mar	19,400	51,256	26.4	44.2	97.0	99.0
April	19,384	70,640	27.1	43.0	97.0	98.9
May	7,094	77,734	22.6	40.6	47.5	89.2
June	0	77,734	-	-	0.0	72.0
July	0	77,734	-	-	0.0	62.7
Aug	9,625	87,359	20.6	34.9	62.7	62.7
Sept	18,098	105,457	25.1	34.6	100.0	67.2
Oct	12,666	118,123	22.2	32.7	73.4	68.2
Nov	17,017	135,140	23.6	33.8	100.0	71.2
Dec	13,369	148,509	23.8	43.8	73.6	71.6
1965						
Jan	15,131	163,640	23.4	25.6	87.1	72.9
Feb	4,407	168,048	14.9	35.0	37.9	70.9
Mar	9,539	177,587	15.3	42.0	72.0	71.9
April	1,850	179,436	12.7	15.8	16.9	68.4
May	8,518	187,955	14.4	37.3	72.8	69.1
June	9,854	197,805	13.3	16.6	100.0	70.9
July	5,445	203,253	12.7	16.9	57.8	70.2
Aug	0	203,253	-	-	0.0	66.4
Sept	8,595	211,848	14.7	29.6	80.8	67.2
Oct	6,451	218,299	13.8	22.5	60.8	66.9
Nov	14,957	233,256	21.2	42.7	98.2	68.3
Dec	16,770	250,026	22.5	28.2	100.0	69.7
1966						
Jan 13	6,890	256,916	9.3	29.7	96.8	70.1

1. Year 1963⁽²⁾⁽³⁾

The Piqua Nuclear Power Facility (PNPF) achieved criticality in June 1963, loading of the fuel was completed in July, and low power testing was conducted through October 1963. The initial power operation commenced early in November. Later in that month, steam-side corrosion led to the failure of two superheater tubes which were repaired by plugging. The reactor was returned to power early in December. On the 28th of December, a shutdown was conducted to initiate drainage of the coolant from the main heat transfer system to permit replacement of the impellers on the primary pumps with units of smaller diameter. By this modification, the primary loop flow rates were reduced from the higher than design values previously occurring.

2. Year 1964⁽⁴⁾⁽⁵⁾

Low power testing was reinitiated on January 8, 1964, and full power operation was reached on January 27. Operation continued satisfactorily through April 6. Following a 10-hr maintenance period on April 6, power operation continued until May 21, at which time a scheduled shutdown was made to conduct: maintenance, scheduled fuel element rearrangement operations, and a scheduled containment building leak test. In the period from April 6 until May 21, several control rod failures due to shorts in the circuitry were reported. During the scheduled shutdown period, commencing May 21, repairs and modifications were made to the control rod circuitry, the in-vessel filters were replaced for the first time, the fuel was rearranged, and control rods were recalibrated.

It was during this latter period that the first of the fuel element inspections was performed. As part of the planned fuel examination program, an evaluation element, Ser. No. P-1071, was removed from core position D-9, See Figure 3, loaded into the shipping cask, and delivered to the Santa Susana Hot Cell facility at Atomics International.*

Before power operations were resumed on August 11, 1964, leaks were discovered in the decay heat removal condenser tube bundle and in the superheater

*Results of the subsequent inspection showed a very thin (estimated as ~0.07 mil) film on the outer heat transfer surface of the fuel cylinder of highest burnup (~1400 Mwd/MTU). Fuel growth was less than anticipated and within acceptable limits.

Figure 3. Core Loading Map

tubes. The tube bundle of the condenser was entirely replaced and the failed tube in the superheater was sealed off. Operation of the PNPf facility was formally transferred from Atomics International to the City of Piqua on August 12. A second decay heat removal condenser tube bundle developed a leak on August 14; a brief shutdown was required for repair.

Power operation continued until October 7, when a shutdown was required to repair another superheater tube leak. On October 15, power was returned until December 7, at which time a shutdown was accomplished to replace the in-vessel filters, as the pressure drop across these filters had reached the Technical Specifications limit of 10 psi.

Fuel element P-1091 in position F-13 was removed for hot cell examinations. This element had shown a steadily increasing outlet temperature over a period of two months.* The media in 15 of the in-vessel filters were replaced. Power

*Subsequent analysis of the element indicated a heat transfer surface maximum film thickness of 1 mil for a peak burnup of 2500 Mwd/MTU (1670 Mwd/MTU average). A "coke" deposit of an estimated 150 gm was found throughout the length of the inner process tube and in the valve. Detailed examination of the element indicated that partial restriction of flow in the inner process tube occurred during the latter operation history.

operations were resumed on December 14 with the filter pressure drop of 8.1 psi. On December 25, the filter pressure drop limit of 10 psi was again approached. One main pump was shut down to reduce flow resulting in reduced pressure drop across the filters (to 5 psi). The reactor operated in this one pump mode until January 28, 1965, during which time the pressure drop increased to 9.6 psi.

3. Year 1965⁽⁶⁾⁽⁷⁾

On January 28, 1965, the reactor was shutdown for the replacement of the in-vessel filters and general plant maintenance. The instrumented fuel element (IFE) was changed from core position E-12 to D-5, a position of lower relative power. The outlet temperature of the IFE had shown an increasing upward trend, indicating a gradual flow reduction through the element.

The reactor was restarted on February 13, and after preliminary testing, brought to power. On February 16, anomalous reactivity behavior was noted during a startup criticality check following a scram initiated by problems with the noise-free electrical system. This was the initial indication of reactivity uncertainties which subsequently triggered an extensive testing and analysis program. The reactor was returned to subcriticality and remained in this condition through February and March except for a series of tests to determine the cause of the reactivity uncertainties, and to measure their magnitude. The test results pointed to a much larger negative power coefficient which was power dependent and a positive pressure coefficient which was power dependent. Analysis performed at that time ascribed this phenomenon most likely to the possibility of organic boiling in the inner process tubes (IPTs) of the highest-power fuel elements at moderate reactor power. Further support for this theory was obtained when physics studies postulating high IPT moderator temperatures showed quantitative agreement with the actual operating observations and, later, a special draining test indicated that the IPTs of a number of fuel elements were either completely blocked or severely restricted. This blockage was believed due to the formation of coke-like masses in the IPT.

During the month of March, several more control rod coil failures occurred as well as one control rod position indicator failure. The reactor was shut down on April 2 to effect repairs of malfunctioning control rod drive circuitry.

During this shutdown, fuel element rearrangement was accomplished and the core loading increased from 61 to 67 fuel elements through the replacement of the six control rod fuel elements. These replacements were undertaken to avoid any possibility that coke-like deposits which had been observed in one of the IPTs might build up to a level that would interfere with proper control rod movement.

The reactor was made critical on April 26, and power and testing operation continued until a scram occurred on May 6. While the reactor was down, the instrumented fuel elements in core positions D-5, D-13, and D-15 were removed to storage. These elements had exhibited increasing cladding temperatures during the preceding operation. In-core fuel rearrangement relocated these elements to lower power positions to reduce their operating temperature. Two new fuel elements and a new instrumented fuel element were added to the core. The reactor was restarted on May 12, and power tests were conducted until May 17. Consistent with power demand requirements, operation took place from May 17 to May 21 to build up the xenon poison in preparation for control rod calibrations. These calibrations took place through May 24. Power operation was resumed on May 25, and continued through June. Several more control rod coil failures were noted throughout this period, including repeated failures (partial in-slippage) on one rod. On June 15, the in-vessel filter pressure drop had increased to 9.9 psi, necessitating shutdown of one main heat transfer pump to reduce flow and thereby lower the pressure drop. The reactor operated in the one-pump mode at an average power of about 13 Mwt until July 18 when the reactor was shut down for maintenance, modification, and in-vessel filter change. During the operation period, Control Rod 5 was considered unusable because of the failure of both raise coils, and the reactor was operated with Rod 5 fully inserted. On July 9, Control Rod 12 malfunctioned. One "hold" coil circuit showed open, and one each of the "grip" and "raise" coils was shorted. The rod was held in full out position by a single hold coil until shutdown. A check of the control rod behavior indicated circuitry problems and corrections to the wiring were made.

The reactor was started up on September 6 and experienced a scram on September 8. Power operation was resumed and continued until October 12, when the reactor was shut down to locate and repair a leak in the superheater tube bundle. A second leaking tube was discovered on October 20 and repaired.

During the shutdown period, the 6 control rod fuel elements of the outer control rod ring were replaced with new fuel elements. Three of the elements had indicated abnormal (high) coolant outlet temperatures and were removed from the core. The other 3 elements were placed in peripheral locations previously occupied by dummy elements. The core was thus expanded to 70 elements. Several control rod coil failures were repaired at this time.

The reactor was returned to operation on October 23 and power resumed on October 24. On October 26, the position indicator coil for Control Rod 11 failed. In compliance with the Technical Specifications, this normally withdrawn rod was fully inserted into the core, resulting in a tilt to the core flux pattern. Modifications to the control rod pattern pointed out another coil failure for Rod 6, but the rod remained operable on alternate coils. Operation of the reactor continued until November 24, when tube failure in the decay heat removal condenser necessitated a shutdown. A coil or a connector failure was also experienced on one control rod. During December, the central control rod could not be raised during a routine rod exercise. Investigation revealed that both sets of raise coils were failed.

3. Year 1966⁽⁷⁾

Operation of the reactor continued from November 24, 1965, until January 13 at an average power level of about 24 Mwt, when it scrammed because of a spurious low degasifier level signal. At time of shutdown, there was no indication of any unusual condition in the reactor core. Control rod drop tests were conducted on January 15. A review of these data did not reveal any significant drop-time increase except possibly for Control Rod 10, which fell in 875 msec as compared with a normal drop time of about 700 msec. Between the 15th and the 21st, a number of control rod circuitry repairs were made. On the 23rd, several rod drop-times were checked again. Rods 1, 3, 4, 7, 11, and 12 had times ranging between the normal 650 and 700 msec. However, during these tests, difficulty was encountered in withdrawing Rod 10 to 41% and, when it was scrammed, it remained at 41%. It was then necessary to drive it in to about 36% before it would drop freely when scrammed. An attempt to drive Rod 9 upward failed to achieve significant movement. Further attempts to obtain normal withdrawal and to drop Rods 9 and 10 were unsuccessful. By switching control rod positions and by visual inspections of the rods, it was established by

January 25 that the failure to drop, which was being observed, was caused by mechanical binding in either the fuel elements or the grid plates.

Attempts to pull the fuel elements from core positions C-8, C-10, and C-12 were unsuccessful with the fuel handling machine. Elements C-8 and C-12 were finally pulled loose by the use of the crane, but could not be fully reseated in the core.

C. PIQUA SHUTDOWN

In view of these difficulties, the top rotating shield was sealed and an investigative program was instigated. The initial action was to unload the fuel from the core; this began on May 2, 1966, and was completed on June 6, 1966. During this procedure, removal force measurements and core probings were made. The probing revealed the presence of a large carbonaceous deposit in the outer moderator region. A mapping of the extent of this deposit was then completed. The deposit was found to extend over the entire central section of the core, radially bound by the outer ring control rods. It was not attached to the upper or lower grid plates but was suspended between the fuel elements, extending from approximately 20 in. below the upper grid plate to 6 in. above the lower grid plate. The volume of the deposit was approximately 4 ft³. A detailed description of the deposit can be found in Reference 8.

Visual inspection of the fuel elements was made at PNPf by using closed-circuit TV and periscope viewing equipment installed in an inspection station.⁽⁹⁾ Buckles were observed to have occurred on a number of elements, particularly those near the periphery of the coke mass. Film deposits on outer process tubes were thicker near the center than near the ends. Shiny-surface finish, outer process tubes showed random deposits, while matte-finish tubes were covered over their entire length. Following this examination, the elements were transferred to the fuel storage pool. Six selected fuel elements were shipped to the AI Hot Lab Facility. The results of the examination of these elements are evaluated in Reference 10.

A more detailed description of the extent of the coke mass and the breakaway forces for the removal of the fuel elements at the fuel element locations can be found in the Safety Evaluation of PNPf Modification.⁽⁸⁾

II. RECOVERY PROGRAM

An engineering effort was undertaken to evaluate the design of the PNPf in light of the operational history of the plant accumulated to January 13, 1966, and the subsequent post-mortem examination of the plant and reactor intervals. An extensive Design Evaluation Report⁽¹¹⁾ was made to the AEC on June 16, 1967. This report fully described these findings, and the engineering analysis results in support of the final conclusion and recommendation for a recovery program with the operation of a restart core using available fuel from Core I operations.

Certain modifications were recommended as a result of the engineering and analysis efforts.

- 1) Modification to the flow pattern in the inner process tubes and the outer moderator regions was required to minimize film formation, and radiolytic and pyrolytic damage to the coolant in the core region.
- 2) Modification and rework of the control drives were required for improved reliability.
- 3)* Redesign of the in-core filter housings was required to allow easier cleaning.
- 4)* Modification of the nitrogen cover gas system was required to assure blanketing of the hot coolant when the core is open.
- 5) Addition of an Attapulugus Clay adsorber column was recommended to improve the coolant purity.
- 6) Modification of the IFE's. Two IFE's were recommended with new top orifices and with expanded temperature measuring capability (8 TC's rather 3) to improve operational surveillance capabilities.
- 7)* Instrumentation was added to allow a direct reading pressure differential across the fuel elements.
- 8)* Holes were drilled (three-3/4-inch diameter) in the outer core panel to prevent potential gas buildup below the upper grid plate.

These modifications will be discussed in greater detail in this section.

Those modifications marked by an asterisk were completed and had been installed or available for installation during the recovery program. The design of the No. 6 modification had been completed. A decision had not been made on the No. 5 modification at the time of program termination.

A further recommendation resulting from the design review was that an intensified surveillance program be established for the restart core to assure a more complete understanding of the restart core operation as it progressed. This is also discussed in Section IV. In addition, the need for further coolant technology studies was apparent after the design evaluation. Consequently, studies were initiated on the oxidation characteristics of the coolant and the thermal decomposition effects of irradiated coolant. These results are covered in Section III.

A. CORE AND COOLANT CLEANUP

The initial phases of the recovery program required that the core, the system piping, and coolant be cleaned to remove the carbonaceous material that resulted from the Core I operation. In addition, it would be demonstrated that the coolant could be cleaned and maintained within specification.

The initial core cleaning operation required draining of the coolant, and removal of the upper core grid plate and the support barrel. The actions utilized for this procedure were as follows.

- 1) Remote viewing techniques and a pool-type manipulator were used. The large pieces of carbonaceous material was grasped and inserted into a remotely held container. Plugs were inserted in the lower grid plate fuel element positions to minimize deposition of matter below the grid plate during the removal operation.
- 2) Initial cleaning, during remote viewing of the vessel, employed a remotely controlled vacuum system to sweep small particulates from the system and inspection of the area below the lower grid plate.
- 3) A final comprehensive remote inspection of the reactor vessel was made to assure removal of the carbonaceous material.
- 4) The inner barrel and grid plate assembly and the freshly charged in-vessel and degasifier filters were replaced, and the system was filled with organic coolant. Coolant temperature was raised with pump heat and coolant was circulated at maximum flow. The operation was continued and the filters were recharged as necessary, until a specified system purity was achieved.
- 5) The heat transfer system was drained, in-vessel filters were removed, and the reactor vessel coolant was drained to permit a final inspection of the vessel.
- 6) All accessible areas inside the reactor vessel were inspected for carbonaceous material. The cleanliness criteria were based on: (1) complete viewing of the core area to ascertain that foreign material did not remain, and (2) verification that each opening in the lower core grid plate was clean.

- 7) The surge tank, the pump inlet screens, the auxiliary building drain tank, and the reactor building drain tank were opened and inspected. All debris was removed. The degasifier tank was not inspected because it had a 6-in. outlet at the bottom of the vessel which was adequate to permit passage of any entrained foreign material. Both upper and lower guide grid plates were inspected at the 100-ft level.
- 8) After approval of the inspection by the plant superintendent, the system was filled with purified organic coolant and circulation and cleanliness were maintained.

The inspection below the lower grid plate (as called for in Step 2) was made with a periscope. Photographs were obtained with a camera attachment.

Circulation of the hot OMP terphenyl coolant at maximum flow was conducted to dislodge and remove any particulates remaining in the reactor vessel and system after the manual cleaning. This phase was started in September 1966. The initial attempt to achieve a high flow rate resulted in an immediate increase in pressure drop across the screens on the suction, and on the two main pumps due to the accumulation of coke on the screens. The screens required cleaning three times during October 1966. Nitrogen was also blown through the Failed Element Location System (FELS) tubes to agitate the contents of the lower portion of the reactor vessel to enhance entrainment and removal of coke. An increase in pressure drop across the in-vessel filters was noted, and the filters in these units were replaced during the third shutdown, for pump screen cleaning. Circulation was then continued through the month of November. Cycling of inlet block valves to produce a varying coolant flow through the vessel and special circulation through the vessel drain line to increase flow velocities under the thermocouple plate were performed in an attempt to dislodge and enhance transport of coke to the pump inlet screens and filters. This phase of the cleanup operation was terminated on December 1, 1966.

Visual inspection of the reactor vessel after draining showed: (1) the presence of a few 1/4-in.-diameter coke pieces in the in-core filter support plugs, (2) the presence of about 50 small pieces of coke^{*} lying on the support plate,

^{*}Some of these pieces later proved to have been improperly identified as coke, and actually consisted of accumulations of organic crystals, or lighting reflections.

the maximum size being about 2 in. on a side, and (3) two pieces of foreign material (a small fitting elbow and a piece of weld bead on the thermocouple support plate). All of the larger pieces of coke were crushed or broken up by use of in-core manipulators and the two pieces of foreign material were removed. Inspection and cleanup of various out-of-core coolant loop components were also initiated with the following results: (1) there was no evidence of coke in either the inlet or outlet liner or other cavities of the boiler, (2) the reactor building drain tank (T-1) contained about 30 gallons of a composite of organic and coke, and (3) the T-2 surge and the T-19 seal drain tank also contained smaller amounts of the same type of materials. These tanks were subsequently emptied and cleaned.

Additional high flow operation with filtering was deemed advisable to complete the removal of the remaining carbonaceous material from the system. The lower core grid plugs were installed in all but 10 core positions in order to maximize the flow velocity over the T/C support plate in the region where the residual coke was observed to be concentrated, and filter support fixtures were installed in the upper core grid plate. Upon initiating flow, an abnormally high pressure drop occurred as a result of plugged filter housings; these filters were then removed, cleaned, and replaced. Coolant circulation was then continued for approximately three weeks in much the same way as before to collect as much carbonaceous material as possible on the degassifier system filters and pump screens. Procedures used, again included special flow paths to agitate the volume below the T/C plate, flow cycling by closing and opening the main pump discharge valves, and nitrogen purging through the FELS tubes. Coolant samples were taken to monitor the progress and effectiveness of cleanup during the operation. The operation was terminated on March 10, 1967.

The inspection after draining showed the following: (1) the plugs in the core lower grid plate had been displaced from their seated positions, (2) the upper core grid plate, the guide grids, and the support barrels were clean of particulate matter but showed the typical black "oil" film, (3) there were two pieces (1-1/2-in. by 1-1/2-in.) of coke observed to be lodged between a thermocouple shroud and a lower grid plate hole, and (4) a small amount of carbonaceous material had accumulated on the main-pump strainers. The two pieces of coke were removed and the pump strainers were cleaned. The total amount

of coke observed in the reactor vessel was estimated to be less than 30 gm and nearly all was subsequently removed from the vessel.

The systems were maintained dry in a nitrogen atmosphere and at ambient temperatures until July 14, 1967. The reactor vessel and the pressurizing and degasification systems were then filled with HB-40. The HB-40 was maintained at a temperature of about 150 to 210°F, and was circulated through the reactor and system periodically by each operating crew (about 1 hr per shift). The HB-40 was then drained and the main heat transfer system filled with the normal OMP coolant. Early in September, 11 downcomer elements, 13 control rod wet test fixtures, 8 special filter support plugs, 53 standard filter support plugs, and the charpy and source holders were installed in the reactor vessel. On September 11, full-flow circulation was established and the coolant temperature was increased to 560°F. This flush was continued until September 20 when the system was cooled and opened, and the filter support plugs and control rod wet test fixtures were relocated to new core positions. Two days later, coolant circulation was resumed at a temperature of 570°F and a total flow rate of 15,400 gal/min. On October 4, this phase of the interim flush was terminated, the system cooled, opened, and the flushing components were removed from the vessel. Special dip-leg and cross-over piping were installed in the vessel to enhance coolant circulation under the thermocouple support plate during the last phase of this flush. On October 6, circulation was reestablished by using these special procedures. During each flush, periodic coolant samples were taken, the flush being terminated when analysis of these samples indicated that the filtration and purification systems had reduced the coolant impurities to a pre-specified criteria level.

The tests used for monitoring the effectiveness of the flushing procedures were the differential pressures across the pump strainers and the degasifier filter, and the radiation levels at these strainers and filters. These measurements proved to be very sensitive to accumulation of carbonaceous material and consequently to the level of carbonaceous material remaining in the system. The coolant samples were analyzed for ash content, particulate counts were made, and Membrane Stain Test (MST) were run. The criteria for demonstrating reactor cleanup were:

Carbonaceous Material (Particulate size greater than 1 micron)	25 ppm (by microscopic counting) max
ASH	5 ppm max
MST	50×10^{-5} A/mg max

The last flush was completed in October and the vessel inspected. No visible pieces of coke were observed in the reactor vessel. The inspection of the upper core grid plate, the lower core grid plate, and the thermocouple support plate was accomplished by use of a periscope at each of the 85 core positions. No carbonaceous material was visible. The beryllium sleeve (source) and charpy element holders were inserted in the vessel during the latter flushing operations. Subsequent inspection indicated that they were clean with no apparent material accumulation. There were no significant changes observed in the monitoring test results (i.e., filter and straining pressure drops and corresponding radiation levels over the period of time of the last flush from 9/21/67 to 10/18/67).

B. CORE AND COOLANT SYSTEM MODIFICATIONS

The requirements for improved hydraulic performance were a result of the engineering efforts during the design evaluation. The modification recommended would result in an increase in inner process tube velocity by a factor of 25 and a factor of about 70 increase in average outer moderator region velocity. As mentioned before, these increased flows would drastically decrease the exposure of the organic coolant to the effects of pyrolytic and radiolytic damage and thus alleviate the problem of film formation and prevent coke formation in the core. The specific changes were based on engineering analysis of hydraulic flow tests performed during the development test program. These tests are described below.

1. Quarter Section Tests

A full-scale geometric mockup of a 90-degree section of the modified core was fabricated and used for water flow tests of alternate downcomer designs. Variations of three basic design approaches for the outer flow modification considered and were tested. In the first, a vertical baffle surrounded by the core formed an annulus between the baffle and core barrel for channeling the moderator flow into the lower plenum. In the second, several fuel element positions, were occupied with various designs of downcomers. In the third, the annulus region of the core was open and did not contain either baffle or downcomers, but discharged to the lower plenum through 24 peripheral holes in the grid plate. The reference design chosen was the spiral orificed downcomers as described in NAA-SR-TDR-12278, "Hydraulic Performance and Testing of PNPf Core 1 Recovery Program with Spiral Orificed Downcomers" C. G. Johnson, December 16, 1966.

**QUARTER SECTION TEST
41 MODIFIED ELEMENT
CONFIGURATION**

LEGEND

- MODIFIED FUEL ELEMENTS
- UNMODIFIED ELEMENTS (DUMMY OR FUEL)
- ⊙ SPIRAL-ORIFICED DOWNCOMERS
- SOURCE ELEMENT

1 to 47 CHANNEL NUMBERS

Figure 4. Quarter Section Test (41 modified element configuration)

It was demonstrated that, with the hydraulic modifications: (1) there were no stagnant areas in the outer moderator region, (2) coolant velocities in the highest flux region were nearly 6 ft/sec (after flow adjustments). A minimum velocity requirement for this region of 2.2 ft/sec, had been established based on studies of projected film formation rates,⁽¹¹⁾ and (3) peripheral annulus region velocities, upward in the lower third of the core and downward in the upper two-thirds, were in excess of 0.5 ft/sec in all cases.

2. Seven-Element Hydraulic Tests

To evaluate the configuration of the redesigned fuel element lower end fitting, Figure 5, water flow tests were conducted on a full-scale mockup of the central portion of the core and the seven central elements. Figure 6 is a schematic diagram of the setup.

The test objectives were as follows:

- 1) Establish the fuel section flow split between the reactor core lower plenum and the moderator region at rated conditions of flow and pressure drop.

Figure 5. Fuel Element Lower End Fitting With Pressure Probes

Figure 6. Seven Element Test Assembly

7736-5103

Figure 7. Modified Fuel Element - Lower End-PNPF Restart Program

7736-5104

- 2) As fuel section flow of a single element is reduced, determine when flow from the inner process tube will pass through the cloverleaf orifice (see Figure 7) in the reverse direction and discharge through the thermocouple flow tubes. This could produce an erroneous outlet T/C reading.
- 3) As flow in the central element is reduced, determine when flow from the surrounding elements at full flow will enter the lower housing of the central element and discharge through the thermocouple flow tubes.
- 4) Evaluate the impact velocity of the IPT discharge flow on the lower housing assembly impact plate.

The results of the test program follow.

- 1) The fuel section flow split was approximately 18% to the reactor core plenum and 82% to the moderator region.
- 2) No inner process tube flow passed through the cloverleaf in the reverse direction and entered the thermocouple flow tubes until element fuel section flow had been reduced to 25% of nominal.
- 3) No flow from surrounding elements entered the lower housing of the central element and discharged through the thermocouple flow tubes until central element flow had been reduced to 10% of nominal.
- 4) The maximum impact velocity of the IPT discharge flow on the lower housing assembly impact plate was less than 22 ft/sec at all IPT flows up to 48 gpm (125% of nominal). This velocity (22 ft/sec) was well below the established maximum impact velocity of 30 ft/sec which was based on erosion rate considerations.⁽¹¹⁾

Figure 8 shows the hydraulic pattern provided in the restart core. The bottom fittings of the fuel elements are to be modified to direct the major portion of the flow to the outer moderator region. Downcomers at the core periphery collect the outer moderator flow and redirect it to the lower plenum.

Coolant flow rates in the restart core fuel elements are contrasted with those of Core I, see Figure 9. Flow rates at the lower end of the modified element are shown in Figure 7. Note the higher inner process tube flow rate and the use of thermocouple flow tubes to direct a portion of the fuel flow to the

DIAGRAM OF FUEL ELEMENT FLOW
 289 REGULAR FUEL ELEMENT
 [302] CONTROL RODDED ELEMENT

Figure 8. Reactor Elevation 41-Element Restart Core

7706-5105

Figure 9. Core I and Restart Core Fuel Elements

A comparison of the operation conditions of Core I and the restart core is given in Table 2.

TABLE 2
PIQUA REACTORE CORE CHARACTERISTICS COMPARISON

Parameter and/or Item	41-Element Restart Core	51-Element Restart Core	61-Element Core I
Power, Rated Full* (Mwt)	30	37.5	45.5
Density, peak/element	1.06	1.13	1.10
average/element	0.73	0.74	0.75
Nominal (at Core I average power) Density,	20.4	20.4	20.4
peak/element	0.72	0.62	0.49
average/element	0.50	0.40	0.34
Flow, System (gpm)	12,500	13,500	13,900
At fuel element, nominal (gpm)	289	258	237
At fuel surface, nominal (gpm)	251	221	235.5
At fuel surface, velocity (fps)	16.3	14.4	15.3
Flow, at Inner Process Tube (gpm)	38	37	1.5
Velocity at inner process tube (fps)	1.7	1.6	0.07
Flow [†] at Outer Moderator (gpm)	10,400	9,300	317
Velocity at outer moderator, nominal (fps)	5.0	4.8	0.07 (down)
Flow at Dummy Element (gpm)	8.3	7.6	3.8
Velocity at dummy element (fps)	0.17	0.15	0.07
Fuel Element Allocation			
Normal elements	27	37 [§]	45-47
Control elements	13	13	13
IFE elements	1	1	3-1

*Shift in peak-to-average power not considered since it varies with control rod position requirements and is always less than 10%.

[†]Total inflow, detailed velocities in text.

[§]Ten unmodified elements used at core periphery.

The 20 dummy elements in the 41-element restart core would fill both the upper and lower grid plate holes, assist in maintaining the desired coolant velocity in the outer moderator region, across the core, and provide locations for additional fuel elements in the core. The dummy elements were redesigned to improve core neutron economy and to increase coolant velocity through the interior of the dummy to inhibit film formation. The dummy element top fitting was the same as that for a fuel element. The tubes connecting the upper and lower fittings were fabricated from 6061-T6 aluminum and were pinned to the end fitting.

The thermal performance was analyzed on the basis of the changed hydraulics. A summary of the predicted clean core thermal operation at 30 Mw is presented in Table 3. In addition, a fouling film analysis was made to determine when the film formation thickness would require repositioning of the fuel elements in order not to exceed the maximum hot channel cladding temperature of 750°F. The results of this analysis are shown in Figure 10 for two of the highest powered elements identified as F-11 and F-9. The figure shows the film thickness and resulting cladding temperature, if the elements are not repositioned, at power levels of 20 and 30 Mw. Film thicknesses and temperatures are plotted as a function of average element exposure. Temperatures shown are for elements F-11 and F-9 operating at maximum power, 1.13 and 1.026 Mw respectively. Repositioning the elements would result in the most efficient operation and maximum utilization of the fuel.

An in-vessel nitrogen purge was designed which would provide uniform gas blanketing over the reactor coolant during fuel handling and maintenance operations. The new header was installed in the reactor vessel just below the upper guide plate. This system will improve the control of oxygen in-leakage by providing additional nitrogen, and the more evenly distributed cover gas will minimize oxygen contamination of the coolant.

The in-vessel core coolant filters design was modified to facilitate maintenance by providing easy disassembly for replacement of filter media, and inspection and cleaning of the inner screen. The ability to thoroughly clean the inner screen was expected to increase the operating life of the filter unit. The same type and size of filter spools used in Core I were to be used in the modified filter element. After a filter element had been removed from the reactor,

Figure 10. Film Thickness at Hot Spot

TABLE 3
CLEAN-CORE, CLEAN-ELEMENT, THERMAL
OPERATION AT 30 Mw

Parameter	Control Element No. F-11	Standard Element No. F-9
Power Generated in Element Fuel (Mw)	1.13	1.026
Average Coolant Velocity in Element Fuel Section (ft/sec)	17.11	16.33
Net Coolant Flow (lb/hr)	118,573	113,633
Effective Film Coefficient (Btu/hr- ft ² -°F)	1,515	1,459
Coolant Inlet Temperature (°F)	542	542
Element Outlet Temperature (°F)	601	598
<u>Hot Channel Factors</u>		
FΔT (coolant)	1.75	1.75
Fθ (film)	1.28	1.28
<u>Cladding Temperature at Hottest Axial Location (°F)</u>		
Nominal	650	644
Hot channel maximum	700	692
<u>Heat Flux (Btu/hr-ft²)</u>		
Average	73,630	66,770
Hot channel maximum	141,400	128,200
Burnout	867,800	849,700

the handling head and inner screen tube could be removed easily. The filter spools could then be removed for maintenance or replacement and the inner screen tube checked for cleanliness and/or cleaned. A fixed orifice was added to the bottom of each in-vessel core filter housing. This orifice was fitted into the top of a fuel element handling head and controls the flow of coolant from the plenum above the upper grid plate into the fuel element. The orifice assured that the fuel elements containing control rods would receive their proportionate share of coolant supply. The section could be replaced when a change of orifice size was required. The amount of material that fit into the fuel element

TABLE 4
CONTROL-ROD DRIVE-POWER COIL MODIFICATIONS

Item	Alternates	Remarks	Selection
Power coils	Original design, but with anodized aluminum wire	Generally unsatisfactory due to low quality anodized aluminum wire; effort would have to be concentrated on inspection technique and quality control.	Reject
	Original design, but with copper wire	Because of form factors involved in insulation, would run hotter than original design; does allow maximum utilization of existing parts. No leakage redundancy between coils.	Utilize as backup modification
	Separately sealed coils with copper wire	Better heat transfer, coils will operate with acceptable temperature rise. Allows for true leakage-redundancy.	Selected as reference modification
Wiring and seals	Wet wiring from power coils to jumper connector (original design)	Generally satisfactory operation, but susceptible to mechanical damage; difficult to repair.	Utilize as backup modification
	Mineral-insulated (MI) sheathed cable from power coils to jumper connector	Very resistant to mechanical damage, allows best seal to power coils.	Select
	Wet jumper cable (original design)	Susceptible to mechanical damage.	Reject
	Wet jumper cable, but protected with flexible conduit.	Protected from mechanical damage, easily movable during fuel-handling operations.	Select
	Dry jumper cable	No advantage over wet design based upon performance. Difficult to maintain seals.	Reject
	Wet connectors at control rod jumper and reactor penetration (original design)	Satisfactory operation in original design.	Select
	All dry wiring and connectors	Would make fuel-handling operations excessively difficult.	Reject
Position coil	Wet wiring (original design) but with stronger upper section	Satisfactory operation in original design. Upper section needs new design to prevent bowing due to thermal stress.	Select
Indicator coil	Dry wiring (original design)	Satisfactory operation in original design.	Select
Thermistor coil	Wet wiring (original design)	Satisfactory operation in original design.	Select
Fixed and movable armatures, drive rod, etc.	Original design	Satisfactory operation in original design.	Select

head was reduced to minimize induced activity and subsequent radiation levels during filter maintenance and orifice changing.

C. CONTROL ROD MODIFICATIONS

The reliability analysis of the PNPf control rod drives were made as part of the design evaluation effort.⁽¹²⁾ Also, the possibility of using ex-core drive units was investigated. The trade-off between the in-core and ex-core drive was between the advantage of maintainability for an ex-core drive vs the difficulty of developing a safe ex-core drive unit. The alternates are discussed further in the "Termination Report - Piqua Control Rod Drive Modification."⁽¹³⁾ With the in-core drive established as the reference design, possible modifications to the control rod drive were examined. Alternates considered for drive components and the basis for their selection are given in Table 4. The selected modification had the following features.

- 1) The power coils were changed from the original unitized design using anodized aluminum wire to a design utilizing individually sealed, truly redundant coils with copper wire insulated with a high temperature ceramic encapsulant.
- 2) The cabling from the control rod jumper plug to the power coils was changed from unprotected wet wiring to mineral insulated (MI) sheathed wiring.
- 3) The upper mechanical section of the position coil was changed to eliminate bowing as a result of earlier design deficiencies.
- 4) The jumper cable from the reactor vessel penetration was changed from an unprotected to a mechanically protected wet wiring cable.
- 5) Other design features which had demonstrated their capability during power operation were retained with no change, thus allowing maximum utilization of existing hardware.

The description of the modified drive can be found in References 8 and 13. A cutaway drawing of the drive is shown in Figure 11.

The new coil winding and fabrication techniques were fully developed, as was the necessary quality assurance program.

7710-2547A

Figure 11. Modified Control Rod Drive

Tests	Test Objective	Poison Section Travel Requirements	Operating Temperature(s) and Cycles	Operating Pressure(s) and Cycles	Transients Above Operating Conditions	Transients Below Operating Conditions	Coolant Analysis	Test Documentation	Test Results
Reactor Design Reference Conditions	-	2140 full-travel cycles, up and down, in 20 yr; based on 100 full-travel cycles, four scheduled test scrams, and 10 unscheduled scrams per year.	350°F refueling to 650°F maximum operating. 100 cycles in 20 yr, two scheduled plus three unscheduled per yr. Rate 50°F/hr.	0 to 135 psig 100 cycles in 20 yr, two scheduled plus three unscheduled per yr. Rate 100 psig/min.	Operating to 750°F and 300 psig. Five cycles in 20 yr. Arbitrary design value. Rates 50°F/hr, 300 psig/min.	400 to 100°F at atmospheric pressure, 10 cycles in 20 yr. Design insertion rate 5 ft/sec arbitrary design value.	Ash 10 ppm maximum. MST value 500 maximum. Carbonyl 600 ppm maximum. Water content 2000 ppm maximum. High boiler 0 to 30%.	-	-
Electrical Feedthrough	Evaluate electrical feedthroughs being considered for CRD modification.	-	300 to 800°F at 83.3°F/hr. 24 cycles.	300 psig.	50 to 300 psig at 300°F. 24 cycles.	-	Santowax OMP.	AI-66-TDR-7.	Single feedthrough satisfactory; multiple feedthrough unsatisfactory.
CRD Bearing	Determine coefficient of friction between armature and bushing for candidate CRD materials.	15,000 cycles.	500°F. No cycling.	-	-	-	Santowax OMP.	NAA-SR-TDR-12320.	300 series stainless steel, with "Tufftride" coating against chrome plate gives best wear surface. Coefficient of friction = 0.69 maximum.
CRD Magnetic Jack Mockup	Verify ability to maintain dimensional alignment during assembly welding and verify ability to withstand handling loads.	-	-	-	-	-	-	TR-691-51-003.	Dimensional stability and ability to withstand handling loads verified.
Mark II (Original Design) Static Dry Test	Establish the static load capability of the Original Piqua CRD and thus provide a basis of comparison for the modified CRD.	-	-	-	-	-	-	DTP-691-51-009.	Modified CRD design shows about 200% better static load capability than original CRD design, as calculated.
Defective Coil	Establish compatibility of the wire insulation system with reactor coolant.	-	1000 hr in 650°F organic. Coil temperature determined by power density.	Saturation pressure corresponding to organic temperature.	-	-	1) Distilled Piqua coolant, 400-ppm H ₂ O initial content. 2) 1800-ppm H ₂ O initial content.	TP-691-51-001. DTP-691-51-001. TR-691-51-001.	Compatibility established, but ETP copper wire embrittled. Change to OFHC copper wire recommended.
Continuation, Defective Coil	Establish compatibility of the wire insulation system with reactor coolant.	-	1000 hr in 650°F organic.	Saturation pressure corresponding to organic temperature.	-	-	Distilled Piqua coolant, 400-ppm H ₂ O initial moisture content.	DTP-691-51-010. TI-691-22-007.	Compatibility established, OFHC copper wire did not exhibit embrittlement.

Table 5. Summary of Control Rod Drive Test Program
(Sheet 1 of 2)

Tests	Test Objective	Poison Section Travel Requirements	Operating Temperature(s) and Cycles	Operating Pressure(s) and Cycles	Transients Above Operating Conditions	Transients Below Operating Conditions	Coolant Analysis	Test Documentation	Test Results
Defective Cable	Determine the performance of a defective MI cable in reactor coolant.	-	1000 hr in organic at 650°F.	Coolant saturation pressure.	-	-	Distilled Piqua coolant, 400-ppm H ₂ O initial moisture content.	DTP-691-51-002. TR-691-22-007.	Satisfactory operation demonstrated for 1500 hr.
MI Cable Qualification	Quality MI cable for PNPF service.	-	100 cycles 350 to 650°F at 100°F/hr-1000 hr at 650°F.	100 cycles 0 to 200 psig at 350°F. 1000 hr at 200 psig. Rate of change 300 psig/min.	10 cycles 650 to 750°F at 300 psig. Rates 100°F/hr, 300 psig/min.	20 cycles 400 to 100°F at atmospheric pressure.	Distilled Piqua coolant and HB-40 as applicable.	ST0403NA0039.	Test not performed due to Program termination.
Effect of Organic Environment on Terminations	Determine effect of reactor coolant on wire terminations.	-	1000 hr in organic at 650°F.	Saturation pressure corresponding to organic temperature.	-	-	Distilled Piqua coolant, 400-ppm H ₂ O initial moisture content.	DTP-691-51-011.	Test not performed due to Program termination.
Coil Thermal Overstress	Determine effect of thermal overstress on power-coil welds.	-	-	Ambient.	60 cycles 400 to 70°F, 30°F/min.	-	Salt bath.	TI-691-51-001.	All welds performed satisfactorily. (No leakage.)
Prototype Magnetic Jack Test	Establish the magnetic, electrical, and thermal performance characteristics of the modified CRD magnetic jack.	-	100 cycles 350 to 650°F at 100°F/hr-1000 hr at 650°F.	100 cycles 0 to 200 psig at 350°F. 1000 hr at 200 psig. Rate of change 300 psig/min.	10 cycles 650 to 750°F at 300 psig. Rates 100°F/hr 300 psig/min.	20 cycles between 400 to 100°F at atmospheric pressure. Rate equivalent to design insertion rate of 5 ft/sec.	Distilled Piqua coolant and HB-40 as applicable.	TP-691-51-004.	Test not performed due to Program termination.
CRD Acceptance	Verify mechanical, electrical, and magnetic acceptability.	-	-	-	-	-	-	ST0620NA0052.	Test not performed due to Program termination.
Lead Production Drive in Reactor	Establish operating levels, determine wear characteristics, provide complete system checkout.	1000 full-travel cycles, up and down, plus 60 scrams.	350 to 550°F, 4 to 10 cycles.	0 to 135 psig, 20 cycles.	-	10 cycles, reactor at 350 to 400°F to control rod thimble in fuel storage pool at 100°F.	Piqua values during test period.	TP-691-51-006. DTP-691-51-006.	Test not performed due to Program termination.
Production Drive in Reactor	Verify that production units attain performance levels established by lead production unit.	12 full-travel cycles plus 10 scrams at 350°F; 12 full-travel cycles plus 10 scrams at 550°F.	350 to 550°F, 2 to 9 cycles.	0 to 135 psig, 8 to 18 cycles.	-	-	Piqua values during test.	TP-691-51-006. DTP-691-51-006.	Test not performed due to Program termination.
Readout Instrumentation Acceptance	Verify acceptability of position system readout equipment.	-	-	-	-	-	-	ST0203NA0068.	Test not performed due to Program termination.

Table 5. Summary of Control Rod Drive Test Program
(Sheet 2 of 2)

Difficulty was experienced in the development of welding techniques. In the coil closure weld between the coil bobbin and coil shell involving 1015 carbon steel weld to Inconel 600, the final solution involved the application of a "butter" or transition material of Hastelloy W, an inert gas welding technique, and a subsequent electron beam welding. The solution of this and other welding problems took longer than originally anticipated. The other critical weld problem slowing production was relative to the weld between the MI cable and the coil case shell. Solutions to the latter problem were still being investigated at the program termination. All welds except the MI-cable-to-shell weld had been successfully developed and qualified.

The summary of the testing program for the control rod drive is presented in Table 5. The results served to confirm design adequacy. Thermal cycling tests on one sample coil indicated that a more than adequate design margin existed. Component tests in a hot organic environment verified the necessity of keeping the coils leak-tight throughout their design life.

The control rod termination report⁽¹³⁾ gives further details. Emphasis was placed on the control of material selection, manufacturing procedures and processes, testing procedures, and inspection requirements.

III. COOLANT TECHNOLOGY

The importance of understanding the mechanisms of film formation on heat transfer surfaces and the possibility of the formation of sludge and coke under the operating environments was demonstrated during Core I operation at PNPf. During the period of design evaluation of the PNPf,⁽¹¹⁾ the need for additional experimental work became apparent. Information was required on the oxidation and pyrolytic effects on the organic coolant.

A. OXIDATION EXPERIMENTS

The primary contributor or precursor to the coolant problem has been identified as the result of oxygen exposure with subsequent contamination of the coolant. Coolant oxidation caused serious corrosion and fouling in the Organic Moderated Reactor Experiment (OMRE). In spite of precautions to avoid it, some oxidation also occurred in the Piqua Nuclear Power Facility (PNPF) as shown by increases in the carbonyl oxygen content following opening of the reactor vessel, and by the substantial oxygen content of some deposits such as film, coke, and sludge.⁽¹⁴⁾ Concerning oxidation rates, it is known that unirradiated coolant (mostly terphenyls) is relatively inert toward oxidation, and that the rate of oxidation increases with increasing concentration of radiolysis products. Of the oxidation products, carbonyl compounds are among the most abundant. Carboxylic acids and phenolic compounds are also formed, and it was shown that carbonyl compounds are converted to carboxylic and phenolic compounds by pyrolysis.⁽¹⁵⁾ In general, there has been little quantitative information on either the rate or the products of coolant oxidation.

A large amount of attention has been devoted to the study of the products and mechanism of the autoxidation (oxidation via molecular oxygen at moderate temperatures) of hydrocarbons. However, most of the work was concerned with autoxidation of alkanes and alkenes at much lower temperatures (<248°F).

Experiments were performed to study the oxidation rates and products under conditions somewhat simulating those in a reactor. The radiation source initially available for this purpose was a cobalt-60 gamma source. The dose rate obtainable with the actual experimental arrangement was not sufficient to produce a significant effect. The organic coolant program was terminated

before any work could be done at higher dose rates, so that the results actually obtained are for pyrolytic oxidation. This proved, however, to be considerably more extensive than had been expected. It was shown that nonterphenyl impurities were responsible for initiating the oxidation. Although the experiments were far from complete when the work was terminated, considerable information was obtained on kinetics and reaction products.

In addition to the oxidation studies, several experiments were carried out to measure the removal of oxygen compounds by Attapulugus clay. While clay adsorption has been tested extensively for coolant purification,^(15,16) its effect on oxygen compounds had not been measured directly.

These oxidation studies were originally a part of the Heavy Water Organic Cooled Reactor (HWOOCR) program.⁽¹⁶⁾ Santowax OM (an ortho-rich mixture of ortho and meta-terphenyls) was selected for study because it was the reference coolant for HWOOCR.

The oxidation experiments were carried out in a small loop in which an argon-oxygen mixture was bubbled through the coolant in a heated reaction vessel. The coolant was stirred by circulating it through an irradiation coil (into which cobalt-60 rods could be inserted), through the clay column, or through a bypass. The absorption of oxygen was followed by monitoring the oxygen content of the effluent gas. Carbon dioxide was also monitored. Samples of gas and of coolant were taken periodically and analyzed for reaction products.

The oxidation was studied at various temperatures (350 to 620°F), oxygen concentrations (1 to 4% O₂ in Ar), gas flow rates (0.15 to 0.66 moles/min), and pump rates (9 to 29 gph). Coolant samples were taken during the experiments and analyzed for oxygen containing compounds.

The oxidation rate of freshly distilled Santowax OM at high temperatures ($\geq 580^\circ\text{F}$) initially increased rapidly to a maximum then slowly decreased by a factor of 3 to 5 to a steady-state oxygen consumption rate (about 20 to 25% of the oxygen available being consumed). It was found that the nonterphenyl impurities present in the distilled Santowax OM caused the peaking in the oxidation rate and also resulted in a tenfold higher steady-state oxygen uptake rate. The activation energy for the oxidation of Santowax OM was calculated at 13 ± 3 kcal/mole. The production rate of oxygenated compounds, including the gaseous products

and bound oxygen compounds in the coolant, appeared to follow the oxygen consumption rate; thus, at high temperatures, the production rates rapidly increased then slowly decreased to a steady-state rate. At high temperatures, a film containing about 4% oxygen was formed in the reaction vessel but not during the low temperature oxidations.

The data also indicated that under the conditions studied the oxygen consumption rate (1) had a first order dependence on oxygen concentration, (2) decreased with increased gas flow, and (3) decreased with decrease in pump rate. An irradiation dose of $\sim 3 \times 10^6$ rad/hr applied to the coolant circulating in the coiled section of the loop had no significant effect on the oxidation rate.

The amount of oxygenated compounds in unoxidized and oxidized Santowax OM was substantially reduced when circulated through Attapulugus clay by means of an absorption process and also by catalyzed decomposition of some of these compounds. This work has been reported in detail in Reference 17.

B. THERMAL DECOMPOSITION EXPERIMENTS

Another difficulty with the organic coolant identified as a result of the design evaluation effort after the shutdown of the PNPf was the pyrolytic decomposition of stagnant or near stagnant coolant. An experiment was conducted to evaluate the decomposition of stagnant, irradiated coolant for up to 11 months at 550 and 750°F. These samples were contained in stainless steel capsules. The capsules were removed from test for analysis of coolant and cover gas composition at the end of 3, 6, and 11 months. The capsule walls were also examined for film deposition. Unirradiated control samples of coolant were similarly treated for 3 months at 550 and 750°F.

Essentially no change in composition of irradiated coolant occurred in samples treated for eleven months at 550°F. The only parameter that did change measurably was the MST result, which increased from 100 to 1000 ($A/mg \times 10^{-5}$) over a period of 11 months.

Considerable decomposition of both irradiated and unirradiated coolant occurred even after three months in test at 750°F, a decrease in total terphenyl from 70 to 10% for irradiated coolant, and a decrease in total terphenyl from 100 to 30% for unirradiated coolant. After six months in test, essentially all the irradiated coolant had been converted to HB, coke, or gas.

The results indicated that the rate constants for the disappearance of terphenyls in irradiated OM are approximately twice those for unirradiated terphenyls. The rate constant for the disappearance of OM as determined in this work agrees with other published results.⁽¹⁸⁾

A similar comparison has been made for HB formation rates. The results do show that for at least the first three months, the HB formation rates for both the unirradiated and irradiated coolants used in these tests are the same.

These tests and the results are more fully described in Reference 19.

IV. SURVEILLANCE PROGRAM

Experience during Core 1 operation indicated that the scope of the surveillance program should be expanded since it had not provided a basis for detecting the existence of the coke formation.

The intensified surveillance program for the PNPF restart core was to assure that the plant would be operated in a safe manner with full knowledge of the plant status and conditions during operation. In particular, it was designed for early detection of film and coke formation in the core.

This surveillance program covered nuclear parameters, control rod performance, thermal and hydraulic performance, coolant quality and other tests. The program detailed information to be monitored and the frequency of monitoring. Table 6 is a summary of this surveillance program. The program also established the operational limits associated with these parameters listed for monitoring that were to be included in the Safety Evaluation and in the PNPF Technical Specifications.

TABLE 6
SURVEILLANCE PROGRAM

Item	Direct Measurement	Data or Information Required	Supplemental Data	Frequency
I. Nuclear Parameters Monitoring				
1. Reactivity Inventory	No	Control rod position and worth, reactor power history, reactor power temperature	None, but predicted reactivity from computer code necessary for difference comparison	Daily on-site; comparison of hourly data by OAP
2. Power Coefficient				
a. Operational Data	No	Control rod position, reactor power	None, but value measured previously necessary for comparison	Twice weekly on-site. Twice daily by OAP(a)
b. Complete Test	No	Specific test-repeat of measurements made during startup program	None, but value measured previously necessary for comparison	Each 1800 $\pm_{-300}^{+200}$ Mwd of operation
3. Pressure Coefficient	No	Control rod position and worth, core pressure change	None, but value measured previously necessary for comparison	Each 1800 $\pm_{-300}^{+200}$ Mwd of operation
4. Isothermal Temperature	No	Specific test-repeat of measurements made during startup program	None, but value measured previously necessary for comparison	Each 1800 $\pm_{-300}^{+200}$ Mwd of operation (b)
5. Dynamic Characteristics ^(c)	No	Test results		Monthly
6. Zero Power Criticality (with zero xenon)	No	Critical rod positions, rod worth, reactor inlet temperature	Fuel burnup records	Each 1800 $\pm_{-300}^{+200}$ Mwd of operation (d)
7. Fuel Burnup	No	Element power history	None	Daily
II. Control Rod Performance				
1. Rod Drop Times	Yes	Drop time measurement	Bi-weekly rod motion tests	At least once in each 90-calendar day interval(e)
2. Rod Motion	Yes	Number of steps/rod travel	Operational rod movement	At least once in each 2-week interval of continuous reactor operation
3. Rod Calibrations	No	Specific Test	Critical rod positions	Each 1800 $\pm_{-300}^{+200}$ Mwd of operation (a)
III. Thermal and Hydraulic Performance				
1. Reduced Flow Test	No	Temperature measurements	None	Monthly for first 3 months of operation, each 90 calendar days thereafter
2. Core Pressure Drop	Yes	Core ΔP , total flow	Coolant loop pressure measurement	Weekly
3. In-Core Filter Pressure Drop	Yes	ΔP measurement	Coolant quality degasifier pressure	Daily
4. Core Pressure and Flow	Yes	Pressure and flow measurements	Loop pressure and flow	Hourly
5. Flow Distribution	No	Reactor outlet, inlet temperature; element power (calculation); element exit TC measurement	Core flow, power, core ΔP	Monthly

Item	Direct Measurement	Data or Information Required	Supplemental Data	Frequency
III. Thermal and Hydraulic Performance (Continued)				
6. Fuel Surface Temperature	No	Reactor inlet temperature; power distribution, element power, element flow (calculated values)	Fuel exit TC data, core flow, core power, core ΔP , element flow calibrations, IFE temperature data	Weekly
7. Instrument Fuel Element(s) (IFE)	Yes	Temperature measurements		Once/shift
IV. Coolant Quality				
1. MST and Radioactivity Analysis	Yes	Test result		5 days/week ^(f)
2. Fe, Cl	Yes	Test result		3 times/week ^(f)
3. Carbonyl Oxygen, Ash	Yes	Test result		Twice weekly ^(f)
4. High Boilers	Yes	Test result		Weekly ^(f)
5. Particle Count	Yes	Test result		Monthly (weekly for 1st month)
6. H ₂ O	Yes	Test result		Monthly
V. Other Tests				
1. Fuel Element	Yes	Visual inspection ^(g)	Fuel element removal and breakaway forces required	Each 1800 ⁺²⁰⁰ Mwd of operation -300
2. Core Probing	Yes	Flow channel condition (h)	None	Each 1800 ⁺²⁰⁰ Mwd of operation -300 (i)
3. Coolant Loop Performance	No	Loop parameters ^(j)	None	Daily, monthly
4. Fuel Management	No	Element power history, relative flow, heat transfer surface conditions	Hot cell examinations of selected elements, (if available)	Weekly

- (a) Nominally, at each power change > 3 Mwt. Monthly data must include at least one set of data over a minimum power change of 15 Mwt at a minimum mean power level of 15 Mwt.
- (b) Also, before and after any significant core loading change. (A "significant core change" is defined as one that results in a measurable change in control rod calibration, power coefficient of reactivity, isothermal temperature coefficient of reactivity, or in the base reactivity level such that the uncertainty in predicting the change is equal to or larger than the statistical uncertainties in the reactivity difference calculations.)
- (c) Reactor transfer function parameters
- (d) Also at each significant core loading change
- (e) Or prior to startup if the 90 days elapse during shutdown period
- (f) During shutdown, with the circulating coolant exposed, the carbonyl oxygen and MST tests shall be performed daily. The other tests can be performed at the option of the plant superintendent.
- (g) One inner ring element
- (h) At least two core positions during each test
- (i) Or in each 6-month calendar interval, whichever occurs first
- (j) Overall performance, also boiler, superheater, and main pumps

V. STATUS AT PROGRAM TERMINATION

A. PLANT STATUS

The flushing of the main heat transfer and degasifier systems was completed in November 1967. The coolant was drained from the system and the reactor vessel inspected. No material was found in the vessel. The vessel was closed and the reactor and primary loop were pressurized to five psig with nitrogen to maintain an inert atmosphere and preventing corrosion. All of the core components (fuel, control rods, dummy fuel elements, filters, etc.) were removed from the vessel. Only the beryllium sleeve and charpy impact specimen holders were to remain in the reactor.

The coolant in the pressurization and degasification loop was drained, transferred to the High Boiler Decay Tank, and then disposed of by burning in the waste fired boiler. The steam side of the boiler and superheater were drained and then filled with nitrogen to prevent corrosion.

The systems were checked for radioactivity readings. A number of tanks were opened and checked to ascertain the degree of contamination for disposal purposes. In general, the levels were low. The interim of the main heat transfer piping had a reading of about 0.3 mrad/hr at 2 inches and the transferable contamination was about 1000 dpm/100 cm².

In accordance with instructions from the AEC, the plant was being maintained in a standby condition until disposition was agreed upon by the AEC and the City of Piqua.

B. FUEL ELEMENT REVISION

The cause of the coke formation in the core was primarily due to an insufficient velocity of coolant through the moderator regions of the core. To correct, it was necessary to redirect most of the coolant flow (after it passed through the fuel element) through the moderator region. This necessitated a redesign of the PNPf fuel element. The redesign included a new inlet (upper head) and outlet (lower head). The revision of the elements required that the existing unirradiated elements be cut apart, inner and outer process tubes and

fuel cylinders be salvaged, and that revised-upper and a new-lower head assemblies be installed. At the time of termination, all of the fuel had been removed and stored. Many of the hardware components had been procured and were in various stages of fabrication. An automatic welding machine capable of welding the process tubes to the upper and lower head was being tested. Upon program termination, all of the parts were packaged and stored pending resolution of the plant status.

C. CONTROL ROD DRIVES

At the time of program termination, parts fabrication, development activity, and the test program were well underway. All engineering drawings and specifications had been issued and were under baseline control. Development activity, with the exception of that associated with the MI cable-to-shell weld, was complete. The test program had reached the point where only design verification testing was to be completed.

The total program interrelationship and status at the time of program termination is described more fully in Reference 13. Two constraints on the program completion are worthy of note. The first constraint was the completion of weld qualification. The difficulty here was in qualifying the MI cable-to-shell weld. This constraint was in turn holding completion of the Prototype Magnetic Jack.

The second constraint was tied to the first: No closure welds would be made to the production magnetic jack coils until the Prototype Magnetic Jack fabrication was complete and the unit had successfully passed dry static and thermal cycling tests. This latter constraint was necessary to avoid any risk of assembling the production CRD's and then finding that rework was necessary.

Figure 12 shows the status of the modified CRD parts at the time of program termination when fabrication was about 46% complete. All parts were protected against corrosion and stored at AI pending the resolution of final disposition.

Figure 12. Status of Modified CRD Parts at Program Termination

AI-AEC-12696
51/52

REFERENCES

1. "Final Safeguards Summary Report for the Piqua Nuclear Power Facility," NAA-SR-5608, August 1, 1961
2. "Piqua Nuclear Facility Reactor Operations Analysis Program Report No. 2 Fy 1963" NAA-SR-8722
3. "Piqua Nuclear Facility Reactor Operations Analysis Program Report No. 3 July 1, 1963 - December 31, 1963" NAA-SR-9473
4. "Piqua Nuclear Facility Reactor Operations Analysis Program Report No. 4 January 1 1964 - June 30, 1964", NAA-SR-10307
5. "Piqua Nuclear Facility Reactor Operations Analysis Program Report No. 5 July 1, 1964 - December 31, 1964" NAA-SR-11142
6. "Piqua Nuclear Facility Reactor Operations Analysis Program Report No. 6 January 1, 1965 - June 30, 1965" NAA-SR-11628
7. "Piqua Nuclear Facility Reactor Operations Analysis Program Report No 7 July 1, 1965 - January 13, 1966" NAA-SR-11995
8. "Safety Evaluation of PNPf Modifications," NAA-SR-Memo-12103, December 27 1966
9. NAA-SR-TDR-12226, "Visual Examination of Piqua Core I Fuel Elements at the PNPf," and "Visual Examination of PNPf Fuel Elements at Piqua" NAA-SR-TDR-11874
10. Evaluation of the Post Irradiation Examination Data from Piqua Fuel Elements Including the Results of P-1114 Examination NAA-SR-TDR-12441 4/25/67
11. Design Evaluation Report (Draft) Transmitted by AI Ltr No. 67AT3415, June 16, 1967
12. "Piqua Control Rod Drive Evaluation," NAA-SR-Memo-12142 Termination Report - Piqua
13. "PNPF Control Rod Modification Termination Report," D. J. Sobo, AI-AEC-TDR-12648
14. "PNPF Reactor Operation Analysis Program, Semiannual Progress Report No. 10, January 11-June 30, 1967, "NAA-SR-12561, (November 1967)
15. R. F. Makens (Editor), "Organic Coolant Summary Report," ID0-11401 (December 1964)
16. "Heavy Water Organic Cooled Reactor Status & Potential June 1967," AI-CE-87 (August 1967)

17. "Oxidation Experiments on Organic Reactor Coolant" Q. R. Stohl and O. T. Tiedeman AI-AEC-TDR-12688 (April 1968)
18. AECL-1915, M. Tomlinson, M. G. Baity, R. R. Tymko, "Radiation Decomposition Characteristics & Physical Properties of Three Low Melting Organic Coolants to Reactors"
19. "Reactor Coking Capsule Experiments on Organic Reactor Coolant," Sullivan, R. J. and Guon, J. AI-AEC-TDR-12689