UNCLASSIFIED

~~OFFICIAL USE ONLY~~

BNWL-766

FAST FLUX
TEST FACILITY
PROJECT REPORT

3-

NPTR HEAT TRANSFER ANALYSIS

Q. L. Baird
H. C. F. Ripfel

April 1968

AEC RESEARCH &
DEVELOPMENT REPORT

~~OFFICIAL USE ONLY~~

UNCLASSIFIED

BNWL-766

BATTELLE NORTHWEST
BATTELLE MEMORIAL INSTITUTE · PACIFIC NORTHWEST LABORATORY
3000 STEVENS DRIVE, P. O. BOX 999, RICHLAND, WASHINGTON 99352

INFORMATION CONCERNING USE OF THIS REPORT

[redacted] STATUS

This [redacted] it is transmitted in advance [redacted] in confidence solely for use [redacted] contracts with the U. S. Atomic Energy Commission. This document [redacted] otherwise disseminated or used for purposes [redacted] before patent approval [redacted] from the Chief, Chicago Patent Group, U. S. Atomic Energy Commission, [redacted] Cass Ave., Argonne, Illinois.

PRELIMINARY REPORT

This report contains information of a preliminary nature prepared in the course of work under Atomic Energy Commission Contract AT(45-1)-1830. This information is subject to correction or modification upon the collection and evaluation of additional data.

LEGAL NOTICE

PACIFIC NORTHWEST LABORATORY
RICHLAND, WASHINGTON
operated by
BATTELLE MEMORIAL INSTITUTE
for the
UNITED STATES ATOMIC ENERGY COMMISSION UNDER CONTRACT AT(45-1)-1830

PRINTED BY/FOR THE U.S. ATOMIC ENERGY COMMISSION

54-1950-016 (11-67)
AEC RL RICHLAND, WASH.

UNCLASSIFIED

[redacted]

BNWL-766

3 3679 00061 3267

NPTR HEAT TRANSFER ANALYSES

April 18, 1968

Prepared by

Q. L. Baird

H. C. F. Ripfel

Approved by

R. H. [illegible] for D. L. Condotta

D. L. Condotta

PACIFIC NORTHWEST LABORATORY
Richland, Washington 99352
Operated by
Battelle Memorial Institute

for the
U. S. Atomic Energy Commission under Contract No. AT(45-1)-1830

[redacted]

UNCLASSIFIED

NPTR HEAT TRANSFER ANALYSES

H. C. F. Ripfel and Q. L. Baird

I. INTRODUCTION

The heat balance and physical design data for the Nuclear Proof Test Reactor (NPTR) are linked together since the heat balance can be investigated only with a sound basis of physical data. A rather important aspect is the isotopic composition of the plutonium in the NPTR fuel. The α-decay heat generated in the fuel has an impact on the safety of the facility. All considerations assume that for ease of operation the NPTR should be maintained at, or close to, room temperature and that design measures (e.g., void introduction) should be such that they minimize the differences between FTR and NPTR.

The following sections include: a consideration of the nuclear and fission heat production(Section II); a physical description of the reference and alternate core features, with special attention to the similarity to the FTR and the necessary accommodations for heating and cooling (Section III); a heat balance determination for the reference core including loss of air coolant considerations (Section IV); a heat balance determination for the molten sodium alternate (Section V); and heating and cooling considerations for the fuel storage and handling area (Section VI). The results are summarized in a conclusion (Section VII).

Cost and operational aspects of the molten sodium alternate feature are discussed briefly in Appendix A. Appendix B is included to propose a high temperature core zone (module).

II. NUCLEAR AND FISSION HEAT

The heat production of the NPTR (core) has been estimated on the basis of plutonium fuel with an isotopic composition such as shown in Table I-1

Table I-1

Fuel Isotopic Composition and Relative Heat Production[1]

Isotope	Wt. Percent	Specific Power
^{238}Pu	0.2%	(550 W/kg)
^{239}Pu	85.8%	(1.8 W/kg)
^{240}Pu	12.0%	(6.9 W/kg)
^{241}Pu	2.0%	(33 W/kg)

For a specific heat generation rate of 4.14 W/kg and a core containing 700 kg of plutonium, the total thermal power is 2893 W (about 3 kW). Slight changes in composition do not affect the heat release appreciably. The maximum short term fission power of the NPTR is 2000 W. Thus, a potential total power of 5 kW exists for the NPTR.

III. Mechanical Aspects of the NPTR Core Design

The overall configuration of the NPTR (split truncated cone core with a gap test zone in the middle) is to match the FTR.[2,3] The interstitial sodium is contained inside the tapered hexagonal module cans (cf. A-0028-R). Minor additional differences are introduced for ease of fabrication of the module cans. There is a radius of curvature on the corners of the cans (0.25 in. at the bottom and 0.75 in. at the top of the module) which introduces void. A tolerance of +0, -0.04 in. is allowed on the width of the modules (across the flats). For the void estimate it was assumed that the tolerance was used to 50%, i.e. -0.02. If this is not realistic, the allowance is to be restricted. Further, there are clearances of 0.005 in. for the exchange of a part (55) of the fuel pins. The clearance for the moving parts of the control and safety rods was estimated to be about 0.02 in. in diameter.

Table III-1 shows the impact of these changes on the void in the core.

Table III-1

Void in the NPTR Core*

	Module Corners		Module Tolerance		Pin - Tube Clearance		
	in^2	v/o	in^2	v/o	in^2	v/o	v/o
Fuel Module	0.08	0.4	0.16	0.81	0.21	1.08	2.29
Control, Safety Rod	0.08	0.4	0.16	0.81	0.10	0.5	1.71
Core Average							2.18 ~ 2.2

*Figures given in in^2-unit volume per unit length, and in volume percent per module in the midplane of the core.

With a nominal total of 76 fuel modules and 18 reactivity-control rods, the average void in the core is 2.2 v/o. As the void is introduced by decreasing the volume originally taken by the sodium, it partly compensates for the higher density of this material at low temperature. This aids in the simulation of the sodium density of the FTR at the nominal zero power operating temperature of 600°F. Notice that the rounded corners of the modules form channels which go all the way through the reactor and are to be utilized as cooling channels.

The reference core provides Al tubes (0.31 in. O.D., .003 in. wall thickness) to facilitate the exchange of fuel pins. This decreases the sodium density as shown in Table III-2 for the midplane of the core.

Table II-2

Relative Material Composition of the Reference NPTR Core Versus the FTR Core*

(voids neglected)

	Fuel		Stainless Steel		Sodium		Aluminum		Total	
	in^2	v/o	in^2	v/o	in^2	v/o	in^2	v/o	in^2	%
NPTR	7.55	(39)	4.98	(25.4)	5.53	(28.2)	1.45	(7.4)	19.62	(100)
FTR	7.66	(39)	4.98	(25.4)	6.98	(35.6)	-	-		
Ratio NPTR/FTR	100%		100%		79%		-			

*Figures given in in^2-unit volume per unit length, and in volume percent per module in the midplane of the core.

A tabulation showing relative volumes at different core levels would reflect the influence of the use of uniform instead of tapered wall thickness on the module can (which affects the SST and the sodium fractions very slightly). The choice of wall thickness will be made on the basis of fabrication aspects. The representation in Table III-2 applies in either case, since the wall thickness will be taken such that duplication is maintained at least in the midplane of the core. The choice of the operating temperature is obviously the lowest temperature practicable. Hence, the sodium will be frozen and the actual temperature level will depend on thermodynamics rather than neutronics considerations.

The alternate concept, which replaces the sodium by NaK, an eutectic composition of sodium and potassium, needs only very brief mention here. Tables III-1 and III-2 apply in this case if the "sodium" is replaced by NaK. The cooling for the NaK concept is less critical (because the mixture is liquid down to room temperature), but requires essentially the same consideration as with the sodium in order to maintain the core at the desired temperature level.

The molten sodium alternate concept would have virtually the same material composition as the FTR (no Al changeout tubes). The major difference would be the void introduction shown in colums one and two of Table III-1.

VI. HEAT BALANCE OF THE REFERENCE CORE (AND ALTERNATE NaK CONCEPT): COOLING

In Section I the heat production of the NPTR core was estimated to be 5 kW in the maximum. With only surface convection cooling of the reactor, the local temperature peaks out at about 620°F. This is considered an undesirably high temperature and may be hazardous, especially since the thermal expansion (of the sodium) causes stress on the can and pins and may result in ruptures. Thus, the need for forced cooling. A safety factor of two in the layout of the cooling system was desired to provide for flexibility in the isotopic composition of the fuel and in the core temperature. Therefore, the cooling considerations are based on an assumed heat production of 10 kW. The building air is used as a cooling medium and is forced directly through the core using the roughly triangular tapered channels at the vertices of the hexagonal modules. The temperature gradient inside the modules is negligible ($< 2\ ^{\circ}F$). Using the Newton formula for heat capacity and an arbitrary temperature increase of 108 ^oF, a coolant flow rate of about 265* ft^3/min is found to be adequate.[4,5] Losses due to flow paths other than coolant channels must be accommodated by the blower system. The heat transfer coefficient shows an ample margin of about a factor of two as compared to the heat production for this situation. The pressure drop was found to be about 1.1 psi with an average air velocity of 92 ft/sec. The coolant air is drawn through the core and released to the stack (or to the building) through filters, thus preventing the spread of contamination in case of fuel cladding failure. Air flow rate, temperature of the inlet and outlet air, and temperatures of at least every third module in the core will be measured and displayed in the control room. The cooling system is on an emergency power supply and some parts such as the blower have a standby backup to be activated in case of mechanical failure. In the case of total cooling system failure, a warning alarm is initiated and the core has to be disassembled. Fuel handling equipment used for this purpose is on an emergency power supply also. (Simultaneous mechanical failure of both cooling system (plus backup units) and handling equipment is not considered a realistic case.) The temperature increase of a representative module without cooling has been estimated to be 6^{o}F/hr on the basis of 5 kW maximum power and the heat capacity of the system[6,7,8], which gives sufficient time for emergency measures, such as unloading the core.

V. HEAT BALANCE OF THE MOLTEN SODIUM ALTERNATE CONCEPT CORE: HEATING

Heating

As mentioned in II, it is possible to operate the NPTR at a temperature above the melting point of sodium. The incentive for this is the avoidance of the sodium density increase (about 2.5%) connected with the phase change (at 208^{o}F), allowing simulation of higher temperatures without introducing greater amounts of void (than the ones shown in Table III-1). In apparent contradiction to statements in III, it may be necessary to provide heating (or at least insulation) to the surface of the reactor to assure that all contained sodium (in the reflector as well as in the reactivity-control rods) remains molten. On the basis of an unfavorable assumption (i.e., that all the grips for handling, support extension rods, etc. act as heat exchange fins) with no insulation (heat exchange in stagnant air) and requiring a temperature level of 250 °F, the heat loss was estimated to be about 18 kW.[5] It appears that a careful design which minimizes the heat losses (insulation) together with electrical heaters installed on the surface of the reactor (and/or in the top of the modules) would ensure the required temperature level at moderate expense. The heating system should be divided into independent circuits in order that malfunction of a circuit could be corrected without disturbing the overall heat balance and temperature distribution appreciably. The temperatures are measured in representative locations and displayed and recorded in the control room. The heating system is on an emergency power supply.

VI. Heating and Cooling Conditions in the Storage Area

A. Cooling

The fuel storage vault consists of a block of concrete of about 150 X 150 X 120 in. which provides space for the fuel modules in 144 round holes of about 6 in. diameter, 6.5 in. apart from one another. The dimensions (of the partition walls) are chosen such that the storage facility is safe against criticality under all conditions.

To provide enough cooling to keep the sodium contained in the modules frozen (i.e., at about 175 ^{O}F), the heat exchange from the modules to the closed concrete vault was found insufficient. Therefore, forced air cooling is applied such as in the reactor. For this purpose the vault will be arranged in a way which provides distribution and collection of the cooling air to and from single storage positions. The geometry of the cooling channels will be such that a rather large cross sectional area is provided for the air flow. Thus, the flow rate required for adequate cooling was determined to be about 700 ft^3/min.[5]

As the heat transfer coefficient does not depend strongly in this geometry on the flow speed (0.66 ft/sec, approaching zero), it is proposed to rely on convection for emergency backup cooling. Provision will be made to bypass the blowers and filters such that the air is free flowing in case of coolant system failure. Flow rate and reference temperatures are measured and displayed in the control room.

B. Heating

If it is desired to heat the modules in the storage vault to maintain the contained sodium in a molten state, i.e., at 250 ^{O}F, only slight modifications are required. Of course, the vault air will only be circulated internally. Either electrical resistance heaters in the individual modules, or a central heater supplying hot air for circulation among the modules, will be used to maintain a suitable temperature. The heat requirements are highly dependent on the degree of insulation of the vault among other factors (such as load), but will probably not be higher than 10 kW.

VI. SUMMARY AND CONCLUSIONS

The physical description of the NPTR core is very little different from that of the FTR (the neutronics aspects are discussed in Support Document 3 of 5) and the physical similarity can be maximized, if required, by use of the alternate concept of molten sodium with no Al changeout tubes. The heat release from the plutonium does not create serious cooling problems and safety measures for handling of the NPTR even under conditions of cooling malfunction are proposed. The NPTR can be operated at elevated temperatures (with the sodium molten at say 250 °F) provided modest heating is applied. The NPTF storage vault cooling (and heating) is provided by air circulation. Convection cooling is possible as a backup to the forced air system.

APPENDIX A

COST AND OPERATIONAL ASPECTS OF THE REFERENCE VERSUS THE MOLTEN SODIUM ALTERNATE

One of the main areas of concern for the molten sodium alternate is the void free insertion of the pins into the molten sodium for loading and exchange. There is, of course, a "wetting" problem which is usually overcome by heating the assembly (3-4 hrs at 600 to 700 ^{o}F). Whereas this procedure is applicable for the first loading and for low frequency exchange of fuel pins also, it may be found too time consuming for high frequency fuel exchange in the NPTR (e.g., once a week or more). Prewetting of pins by heating them under sodium before insertion into the module cans, would obviate further heat treatment of the modules. The proper choice of operating temperature and the sodium purity required are not clear cut. These are areas where further investigations are planned. Some concern also exists as to the galling of the pins during insertion into the molten sodium as well as into the Al changeout tubes. Coating of the pins with teflon (of similar fluorocarbon) is suggested.

Cost estimates show that there are appreciable fabrication savings (in the order of $30,000) if the modules are not provided with aluminum changeout tubes. These savings would contribute toward covering additional costs for heating and fuel-exchange equipment incurred in the case of the molten sodium alternate.

APPENDIX B

A HIGH TEMPERATURE NPTR TEST MODULE

A high temperature NPTR Test Module is considered a very valuable tool for research in the following major areas; causes of fuel failure (including vibration, hot spots, coolant blockage), early detection of effects that lead to fuel failure, and detector development. Many detectors are in principle available and their functions are fully understood, however, their validity for surveillance of a system like a fast power reactor is far from complete verification. Combinations of the outputs of such detectors as thermocouples, flow meters, neutron flux detectors and their evaluation by correlation analysis are required to yield specific information on function (or malfunction) of an individual reactor element. A high temperature module may also (at least locally) simulate FTR power operating temperature.

A very preliminary design concept investigation shows reasonable feasibility. The high temperature module as proposed here would contain stagnant sodium - in contrast to a "full flow sodium loop". An electric heating system and "super" insulation (multiple foil wrap to reduce radiation losses) on the surface should be sufficient to attain at least FTR power operating temperature. For these reasons it would be available at modest cost.

REFERENCES

1. Conceptual Component Design Description for the Reactor Core (System #31), A-0036, November 29, 1967.

2. E. R. Astley, Progress Report, Fast Flux Test Facility, Reference Concept, BNWL-470, August 1967.

3. Q. L. Baird, R. A. Bennett, C. W. Higby, and P. L. Hofmann, Supplementary Information for FTR/NPTF Evaluation, BNWL-559, September 1967.

4. K. Nesselman, Angewandte Thermodynamik, Springer, 1950.

5. Dubbels Taschenbuch fur den Maschinenbau, Springer, 1950.

6. R. C. Weast and S. M. Selby, Eds., Handbook of Chemistry and Physics, 48th Edition, 1967.

7. S. McLain and J. H. Martens, Eds., Reactor Handbood, 2nd Edition, Volume VI, 1964.

8. H. Etherington, Ed., Nuclear Engineering Handbook, 1st Edition, 1958.

30 U. S. Atomic Energy Commission
Division of Reactor Dev & Tech
Washington, D. C. 20545
M Shaw, Director, RDT
Asst Dir for Nuclear Safety
Analysis & Evaluation Br, RDT:NS
Environmental & Sanitary Engrg Br, RDT:NS
Research & Development Br, RDT:NS
Asst Dir for Plant Engrg, RDT
Applications & Facilities Br, RDT:PE
Components Br, RDT:PE
Instrumentation & Control Br, RDT:PE
Systems Engineering Br, RDT:PE
Asst Dir for Program Analysis, RDT
Asst Dir for Project Mgmt, RDT
Liquid Metals Projects Br, RDT:PM
FFTF Project Manager, RDT:PM(3)
Asst Dir for Reactor Engrg
Control Mechanisms Br, RDT:RE
Core Design Br, RDT:RE (2)
Fuel Fabrications Br, RDT:RE
Fuel Handling Br, RDT:RE
Reactor Vessels Br, RDT:RE
Asst Dir for Reactor Tech
Chemistry & Chemical Separations Br, RDT:RT (2)
Fuels & Materials Br, RDT:RT
Reactor Physics Br, RDT:RT
Special Technology Br, RDT:RT
Asst Dir for Engrg Standards

1 AEC Idaho Operations Office
Nuclear Technology Division
CW Bills, Director
P. O. Box 2108
Idaho Falls, Idaho 83101

3 AEC Richland Operations Office
FFTF Program
JM Shivley
Federal Building
Richland, Washington 99352

1 San Francisco Operations Office
Director, Reactor Division
2111 Bancroft Way
Berkeley, California 94704

4 AEC, RDT Site Representative
Pacific Northwest Laboratories
PG Holsted (2)
LR Lucas
AD Toth
P. O. Box 550
Richland, Washington 99352

4 AEC, RDT Site Representatives
Atomics International
P. O. Box 591
Canoga Park, Calif. 91305

Argonne National Laboratory
9700 South Cass Avenue
Argonne, Illinois 60439

Atomic Power Development Asso
1911 First Street
Detroit, Michigan 48226

General Electric Co.
310 DeGuigne Drive
Sunnyvale, Calif. 94086

1 Argonne National Laboratory
LMFBR Program Office
9700 South Cass Avenue
Argonne, Illinois 60439

43 Battelle-Northwest

SO Arneson
ER Astley
QL Baird
JM Batch - 328
RA Bennett
DC Boyd
CL Brown - 209E
CA Burgess
BA Chandler
TT Claudson - 326
PD Cohn - 3702
DL Condotta
DR Doman
RV Dulin
SL Engstrom
LM Finch
VW Gustafson - FB
JP Hale
KM Harmon
RA Harvey
BR Hayward - 308
RJ Hennig
GM Hesson
CW Higby
PL Hofmann
EM Johnston
DD Lanning - ROB
CW Lindenmeier
WW Little
JW Mitchell
CA Munro
RE Peterson
HCF Ripfel
WE Roake - 308
DW Shannon (2) - 324
DE Simpson
MO Slater
DD Stepnewski
KG Toyoda
RC Walker
FW Woodfield
FFTF File (2)
Tech Info (5)