LICENSE POWER CAPACITY OF THE PUR-1 RESEARCH REACTOR

by

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A Thesis

Submitted to the Faculty of Purdue University

In Partial Fulfillment of the Requirements for the degree of

Master of Science in Nuclear Engineering

School of Nuclear Engineering

West Lafayette, Indiana

December 2018

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ACKNOWLEDGMENTS

I would like to first thank Dr. Robert Bean for trusting me with the care and management of the PUR-1 facility. The day-to-day management, licensing, and public outreach which goes on around the facility has taken my engineering life from practice to profession while developing my ability to speak to a wide variety of audiences.

Additionally, I thank David Storz for his steady hand of maintenance and daily tasks. David's diligent work alleviates much of the strain and allowed me to pursue other facility goals.

To Dr. Chan Choi and Dr. David Chichester, I thank you for asking all the right questions. Taking the fusion course load from Dr. Choi, challenged my technical and mathematical skills. During my summers spent at the Idaho National Laboratory, Dr. Chichester quickly and adeptly honed my ability to work within a research group and identify key areas of research questions.

Finally, I thank Marguerite Bair, her sister Alberta Bair and the Charles M. Bair Memorial Trust for their funding of my undergraduate education. Without their charitable scholarship, attendance at a wonderful school such as Purdue University would have never been possible.

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ABSTRACT

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Institution: Purdue University
Degree Received: December 2018
Title: Licensable Power Capacity of the PUR-1 Research Reactor.
Committee Chair: Robert Bean

This work aims to develop a theoretical power operations envelope for the PUR-1 reactor. Given the bulk coolant temperature, the reactor's power level is limited primarily by the Onset of Nucleate Boiling. Additional limitations to the reactor power are explored including the dose rate at the top of the pool due to shine and the airborne effluent of argon and nitrogen. Operations in excess of the facility cooling capacity will be proposed and are already permitted at other US research reactor facilities, provided temperature limitations are met. The MCNP and NATCON code packages have been implemented to assist in power limitation measurement. A brief discussion on the licensing considerations is included to provide some framework for pursuit of these higher power levels. The maximum power consideration ensures continued full use of the facility while maximizing its effectiveness in the teaching laboratories and access to researchers. The final power level is limited by the administrative dose limit at the top of the reactor pool as well as the Onset of Nucleate Boiling power level as a function of bulk pool temperature. The result is an operational envelope which would allow operators to have the maximum neutron flux without changing the facility or creating phase transition within the light water coolant.

1. INTRODUCTION

1.1 Research and Test Reactors

The nation's fleet of Test, Research, and Teaching Reactors (TRTRs) is a valuable resource for the evaluation of nuclear materials, inspection of samples, and the preparation of the future nuclear work force. These reactors are at a significantly reduced power level than their industrial counterparts but are much more available to researchers and scientists looking to quickly and effectively evaluate early work. Their reduced power level yields a small risk portfolio and allows for maximum flexibility. However, this low risk must be countered with a power level sufficient to perform research in a radiation environment valued by customers.

Currently licensed research reactors in the United States range in power from 100 Watts at the Rennselaer Polytechnic Institute[1] to 10 MW at the University of Missouri-Columbia[2]. This power range covers many orders of magnitude and represents the very diverse nature of the fleet. The Purdue reactor currently operates at 10 kW and is at the lower end of the spectrum. Many of the University research reactors fall in the hundreds of kilowatts to one megawatt range.

Universities are the principle operators of the small reactors across the United States. The reactors support both basic research and educational missions. There are currently 25 reactors located at 24 different universities across the US and the location of the reactors are not necessarily correlated with states who have a large industrial reactor portfolio. Indiana, for example, has no commercial nuclear power plants but houses the PUR-1 at Purdue University.[3] Following successful reactor builds at Idaho National Laboratory, formerly Argonne West, many reactors were built across the country. During this time period (1950-1980), many believed nuclear power could be the major electrical production method in the United States. However, with events at Chernobyl and Three Mile Island, the nuclear dream began to fade, and with it, the number of research reactors. The number of reactors on university campuses has declined significantly since a peak several decades ago.

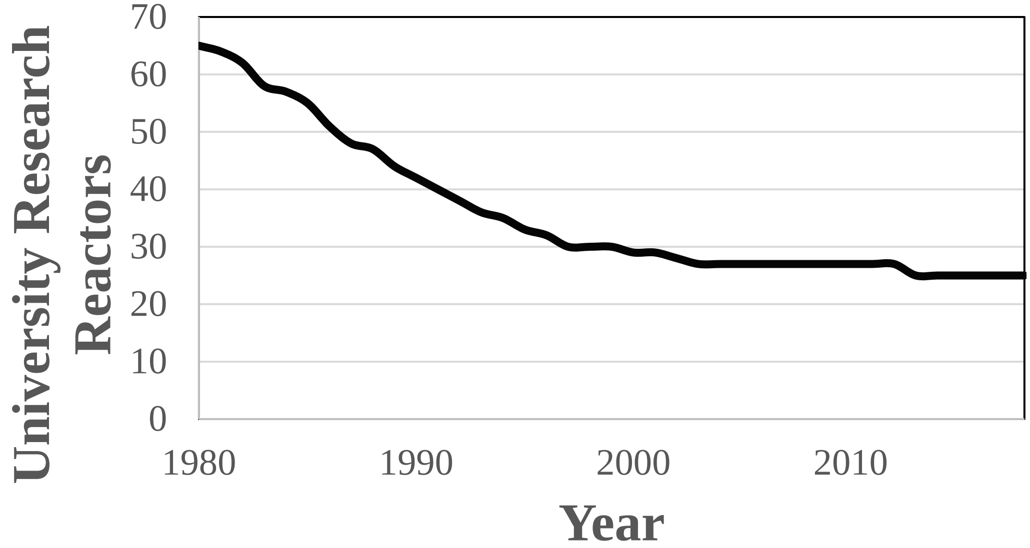

Figure 1: Declining number of research reactors since 1980. [4]

These reactors shut down for a variety of reasons generally attributed to a combination of a lack of adequate funding, loss of public support, loss of institutional support, burdensome regulation, and failure of implementation of aging management strategies.

1.2 The Case for Research Reactors

While the number of reactors located at universities has declined, the demand for a skilled nuclear work force has not. The United States' electrical production from nuclear power has remained at a near constant of 20% over the previous several decades while the population and electrical demand of that population continues to rise.[5] Gains in capacity factor, operational efficiency, and limited outages have enabled nuclear power to maintain its representative proportion of production within the US. Many of these gains are also attributed to continued development of a strong nuclear work force who recently has been working on "Delivering the Nuclear Promise®". This is a strategic initiative led by the Nuclear Energy Institute to continue to increase plant efficiency, reliability, and safety.[6]

The students who have operational time at a reactor facility can make a clear case for improved readiness for integration into future occupations. Approximately 650 Bachelor of Science degrees were awarded in Nuclear Engineering in 2015. Of those degrees, roughly 60% were given at an institution with a nuclear reactor.[7] The US Department of Energy also recognizes the value of a research reactor at higher education institutions. The DOE-NEUP Infrastructure grant program has been a steady and continued force in the maintenance and upkeep of reactors in the country by supplying millions of dollars to these facilities.

1.3 Purdue University Reactor Number One

The Purdue University Reactor Number One is a small research reactor located in West Lafayette, Indiana. The reactor has seen transformational change over the previous decade through completion of a license renewal, power uprate, replacement of control console, and complete overhaul of the regulatory compliance system. The replacement of the Instrumentation and Controls (I&C) is expected to be completed in the Fall of 2018. The text, except where otherwise noted, assumes this change has already been accepted, licensed by the NRC and fully implemented in order to maximize relevance for future readers.

PUR-1 was originally conceived in 1960 as a much larger facility to be located on the Wabash River. Due to the flooding trends of the river and its seasonal unpredictability, a smaller reactor was chosen for the initial construction, which would allow the builders to prepare for more reactors in the future. The facility was built over the course of a two-year timeframe and ultimately went critical in August of 1962. Early work included the characterization of the neutron flux, measurements of the natural convection flow rate, calorimetric and activation methods of power calibration, and other fundamental research. As time progressed, the facility shifted towards radiation exposure experiments.

The primary purpose the reactor today is to be an elite teaching and training laboratory for future nuclear engineers and other stakeholders in the nuclear community. Given its easily accessible location on the main Purdue campus as well as the extraordinarily low risk profile of the facility, operators in training can receive hands on experience without the financial burden of potential missteps at industrial facilities. The secondary mission of the facility is to provide researchers with a tool to perform neutron activation analysis, benchmark reactor codes, and perform low level material tests.

1.4 Reactor Power Uprate Overview and Work

The work described herein is to maximize the operational flux of the PUR-1 reactor. This maximum flux has historically been a static value which is calculated given an estimated instrument uncertainty, coolant temperature, core configuration, reactor room volume, room exhaust rate and many other factors. However, these values are overly conservative when the facility is in certain states. For example, at initial start-up, the reactor's coolant is at room temperature and the Onset of Nucleate Boiling Ratio is far from unity. After the reactor has been operating, the coolant's temperature continues to rise and, given a steady reactor power, this ratio has changed. This work aims to take the dynamic nature of the facility into account and provide a more flexible operations envelope.

No modifications to the facility are proposed. This work solely considers the as-built design. Therefore, limitations on the duration at which the reactor power exceeds the cooling capacity by the heat exchanger are determined by the excess power and the bulk coolant temperature. Additionally, the reactor's radiation dose to a worker is also considered with the current level of shielding and the worker located at the maximally exposed location outside the coolant volume. While some changes to the facility would yield growth to the operational envelope, these changes are outside the budget of the small facility and therefore left for future work.

The result of the work is a proposed power envelope. This will give a range of powers at which the facility can safely operate without exceeding the limitations in the Technical Specifications, Safety Analysis Report, or stated constraints. Continuous operation will require continuous evaluation of the reactor state and its position within the envelope. For example, if the reactor is near the Onset of Nucleate Boiling temperature, the energy addition to the bulk coolant volume will further increase the temperature and lower the margin to boiling. The reactor operator must then respond by lowering the reactor power, creating a slower increase in temperature. This feedback between reactor power and coolant temperature would continue until the reactor power has reached the maximum cooling capacity of the heat exchanger and there is no longer a temperature rise within the coolant volume.

2. PUR-1 REACTOR LICENSE

2.1 Licensing Document Overview

There are several important documents in the licensing of TRTRs. These include the License itself, Technical Specifications, Safety Analysis Report, Safety Evaluation Report, and Environmental Impact Statement. The license is issued by the NRC and details the correspondence between the licensee and the regulator which has led to the issuance of the document. Ultimately, all regulations, guidance, and supporting documents trace themselves back to the Atomic Energy Act of 1954 and its subsequent amendments. Research reactors fall under a variety of regulatory spaces depending on their functionality and power level. These classifications are outlined in 10 CFR Part 50.21. The PUR-1 is licensed as a "104c" facility: "A production or utilization facility, which is useful in the conduct of research and development activities…". [8]

The Safety Analysis Report (SAR) is an applicant (future licensee) prepared document which outlines the safety and operational basis for the reactor. Nearly all modern TRTR SARs are written to comply with the nuclear regulatory guidance (NUREG) 1537 – "Guidelines for Preparing and Reviewing Applications For the Licensing of Non-Power Reactors." These SARs have up to 16 Chapters:

1. The Facility
2. Site Characteristics
3. Design of Structures Systems and Components
4. Reactor Description
5. Reactor Coolant Systems
6. Engineering Safety Features
7. Instrument and Control Systems
8. Electrical Power Systems
9. Auxiliary Systems
10. Experimental Facilities and Utilization
11. Radiation Protection Program and Waste Management
12. Conduct of Operations

13. Accident Analyses
14. Technical Specifications
15. Financial Qualifications
16. Other License Considerations.

Throughout the document, the applicant outlines their case for the safe operation of the facility as well as their ability to decommission once that time arrives. [9]

The Safety Evaluation Report (SER) is prepared by the NRC or their selected subcontractor. It is an evaluation and review of the SAR as submitted. The guidelines for this review are outlined in Part 2 of NUREG 1537. The stated purpose of the standard review and acceptance criteria are to "ensure the quality and uniformity of reviews by presenting a definitive base from which to evaluate applications for license or license renewal." [9]

The Technical Specifications (TS) are built from the SAR and are designed such that when they are followed, the fundamental assumptions of the SAR remain true and the safety conclusions derived therein are maintained. The TS generally follow the accepted guidance of ANS/ANSI 15.4 – "The Development of Technical Specifications for Research Reactors."[10] The TS generally have six chapters:

1. Definitions
2. Safety Limit and Limiting Safety System Setting
3. Limiting Conditions for Operation
4. Surveillance Requirements
5. Design Features
6. Administrative Controls

The Limiting Conditions of Operation (LCO) lay out the requirements which must be met prior to operation while the Surveillance Requirements outline how the facility must maintain the reliability of the LCO instruments such that they can perform their safety functions.

The described work in the subsequent sections show how changes to the Technical Specifications to allow for a higher operational power, based on coolant temperature and radiation at the pool top, could be used to bolster the portfolio of the PUR-1. The changes outlined affect both the Safety Analysis Report and the Technical Specifications.

2.2 Outcome of Power Envelope and Changes To Facility Documents

The addition of a dynamic power level limit to the PUR-1 reactor presents a challenging license scenario. During traditional NRC inspections of operations, spot checks are performed in the reactor logs to determine adequate compliance with the Technical Specifications. The spot checks would, following completion of the proposed change, require a log of all limiting conditions continuously. With the new digital instrumentation installed at the facility, these parameters are readily available. An inspector would be required to determine the radiation level, the coolant temperature, and the reactor power level at any given time and verify they fall under the proposed operation curve.

2.2.1 Limiting Safety System Setting Change

Section 2.2 of the Technical Specifications are titled, “Limiting Safety System Setting.” The current specification is a single line which reads, “The measured value of the power level scram shall be no higher than 12.0 kW.” This line would be struck out and read, “The measured value of the power level scram shall fall within the operational envelope as defined by the limiting chart shown below:”. Note: this proposed chart represents the final conclusion and presentation of this work.

As noted above, the reactor’s time of operation would be reduced while operating at power levels beyond which the heat exchanger is capable of providing. However, this capability exists in the current reactor license. The power uncertainty for the analog equipment was 50%. For a power level of 12.0 kW, this implies this the reactor power level could be as high as 18 kW. However, the heat exchanger is only rated to 10.55 kW giving a certain coolant temperature rise over time. With the proposed work, this coolant temperature rise rate is simply increased.

3. PURDUE UNIVERSITY REACTOR NUMBER ONE

This section will highlight the features of the Safety Analysis Report for the PUR-1 such that a reader is more familiar with operations, characteristics and other considerations of the facility.[11]

3.1 The Facility

The PUR-1 is a small research reactor located in the basement of the Electrical Engineering Building on Purdue's West Lafayette campus. Originally constructed in 1962, the reactor was designed to operate at a nominal power level of 10 kW. However, the original license submitted to the NRC was for a tenth of this value. The power level was re-evaluated during the 2008 license renewal and approved for an uprate to the designed 10 kW in the Fall of 2016.

The reactor is housed inside a dedicated reactor bay. This room contains the reactor control system, reactor pool, temperature and humidity control system, water process system and all associated radiation monitoring equipment. There are no windows to the outside and the floor of the room sits approximately 10 feet below ground level.

The reactor tank has its base an additional 13 feet below floor level. Its total depth is 17 feet giving approximately four feet of water above the ground. The tank has a capacity of 6400 gallons of water. The water is circulated through a water process system at a rate, during operation, of at least five gallons per minute.

The operator control console sits approximately 10 feet from the edge of the tank and consists of two operator screens, an annunciator panel, a suite of neutron flux monitoring units, and other supporting equipment. Operations require at least one NRC licensed operator and a second person at all times.

As discussed, this work proposes no changes to the physical design of the facility.

3.2 Site Characteristics

Tippecanoe County is covered by glacial drift to depths up to 300 feet. The bedrock underneath is of the Mississippian period and consists of limestone, sandstone, flint and shale. The university and the Electrical Engineering building sit above a large deposit of sand and gravel at an elevation of approximately 700 feet. The Wabash River, which regularly floods the surrounding area is well below this level at 500 feet. The nearest active seismological area is the Wabash Valley Fault system of southern Indiana. The Greater Lafayette area is a moderately populated city of approximately 150,000 people.

The proposed licensing change would have no effect on the site characteristics.

3.3 Design of Structures, Systems, and Components

The control system and the reactor are designed to take moderate amounts of seismic damage and small amounts of meteorological damage. The reactor core itself sits some 25 feet below ground level protecting it from common natural disasters in Indiana such as tornadoes. Potential flooding events could cause significant damage to the control system but do not pose a safety hazard to the core itself.

The change to a dynamic operating power level would not affect the design of any structure, system or component.

3.4 Reactor Description

The reactor is composed of up to 208 low enriched fuel plates arranged in 16 assemblies. The assemblies are elevated two feet off the floor of the pool and have inlet nozzles on their lower end to allow coolant entrance. Coolant from the 6400 gallon tank enters the assemblies from below, is heated throughout the assembly and exits to the bulk pool volume above. The reactor's control rods are made of borated stainless steel, for the two shim safeties, and regular stainless steel for the regulating rod. The core is moderated (and cooled) by the bulk light water volume. The core is reflected by a set of graphite assemblies and the light water. A plutonium beryllium neutron source is implemented to promote criticality safety, lessen start-up time and provide some physical phenomena for teaching purposes. The reactor pool is surrounded by 18 inches of concrete and the top is open to the reactor bay air.

While the power level envelope would change the coolant temperature rise rate, there would be no changes to the coolant's volume, reactor loading, or other intrinsic characteristics.

3.5 Reactor Coolant Systems

The reactor's light water coolant is continuously circulated through a primary coolant loop. This loop features a filter, demineralizer, flow rate meter, heat exchanger and pump. The flow rate is required to be at least five gallons per minute. The heat exchanger is shell and tube type and rejects heat to the city water supply where it is drained. There is no nitrogen or argon control system for limiting effluent to the reactor bay air.

The larger power capabilities of the reactor would likely increase the utilization of the reactor coolant systems but would not require a change to the design or implementation itself.

3.6 Engineering Safety Features

The primary safety features of the PUR-1 reactor are the cladding and the confinement. The aluminum cladding on the fuel prevents fission fragments from being released. Maintaining the coolant water chemistry promotes clad integrity. The reactor bay confinement keeps air within the space to a value of negative 0.05 inches of water column. This negative air pressure prevents leakage of argon effluent or airborne radioactive contaminants from entering the rest of the laboratory and the Electrical Engineering building as a whole. The main air inlet and outlet are protected with HEPA filters and the drain contains a HEPA filtered inverted opening as well. The PUR-1 has no containment vessel or emergency core cooling system.

There are no proposed changes, additions or modifications to the Engineering Safety Features of the facility to implement a power operations envelope rather than the static operational power level currently implemented.

3.7 Instrumentation and Control Systems

The Instrumentation and Control of the PUR-1 feature a fully digital suite of Mirion Technologies neutron flux monitoring systems as well as a digital control interface. Operators may move any of the three control rods, the fission chamber, or the neutron start-up source from the control console. Measurements of water chemistry, room air pressure, pool temperature, and dose rate are available at the control console. The four neutron flux monitors protect against high power and high reactor change rate through the use of a fission chamber, compensated ionization chamber, and two uncompensated ionization chambers.

The current instrumentation and control systems of the PUR-1 have capabilities far beyond their current utilization. If there were concern about their ability to handle the increased flux, their location with respect to the core could be changed to produce a lower incident flux.

3.8 Electrical Power Systems

The electrical power for the facility is provided through the building's power system. As power is supplied, it is regulated by two Uninterruptible Power Supply (UPS) units. These contain batteries that allow for at least 30 minutes of continued operation without interrupting normal operations. Normal facility procedure is to immediately begin a controlled shutdown on loss of building power.

There are no proposed changes to the electrical power systems for a dynamic operations envelope.

3.9 Auxiliary Systems

The air quality in the reactor bay is carefully controlled by an industrial grade HVAC system. The HVAC unit controls the room temperature through air conditioning or a warm water supply. It also manages the room humidity as the pool evaporates approximately 35 gallons of water per week. The room also has a fire protection system. The implementation uses a gas filled line that requires both a heat sensor and smoke detection for water activation. Standard communication methods for normal operation and emergency are available in the room in the form of a land line and cell phone reception. Finally, as an auxiliary system, spent fuel storage racks are located within the main reactor pool on the opposite side from the core.

The auxiliary system most affected by an increase in reactor power would be the increase in pool evaporation. With a higher average temperature of the coolant, there would be an added requirement to maintain coolant volume. Automatic coolant addition is currently available but not used at the facility.

3.10 Experiment Facilities and Utilization

Very simple experiment facilities are available for use at the PUR-1. Dry drop tubes made of PVC pipe run along the south of the reactor. Experimenters are able to prepare their sample, attach a retrieval string, and drop it down the tubes. There are also irradiation locations within the west side of the graphite reflector for longer termed experiments. Additionally, an experiment can be placed in the coolant in close proximity to the core.

The driving force behind this work is to increase the reactor flux and thereby the utilization of the experimental facilities. This being noted, there is no required change to the experimental facilities for the enhanced flux.

3.11 Radiation Protection and Waste Management

Radioactive products are rarely produced at the PUR-1 outside of those experiments placed down the irradiation tubes for neutron activation analysis. Argon production will be discussed with more detail in subsequent sections.

3.12 Conduct of Operations

The PUR-1 reactor facility follows a standard operational structure and staffing levels for a smaller research reactor. The Level 1 is a higher ranking University official to whom the Laboratory Director (Level 2) person reports. The Reactor Supervisor (Level 3) performs much of the day to day responsibilities and is assisted by reactor operations staff such as an electronics technician. Future operations at the PUR-1 will likely feature undergraduate and graduate students as well.

With this proposed work, there are no changes to the outline of the operations of the reactor facility however methods for audits would have to be carefully reviewed.

3.13 Accident Analysis

Research reactors generally consider a wide range of accident initiating events including loss of coolant accident, ramped and prompt reactivity insertions, fuel handling accidents, fire, acts of sabotage and experiment malfunctions. Due to the safety related nature of these calculations, they are not discussed in detail here.

3.14 Technical Specifications

The Technical Specifications outline the Safety Limit (LSSS) for the reactor (maximum fuel temperature of $530°C$), the Limiting Safety System Setting (maximum power level), the Limiting Conditions for Operation, Surveillances, Facility Design, and Administrative Responsibilities. They consist of six separate sections and are a separate document from the Safety Analysis Report. Proper execution of the Technical Specifications ensures the assumptions in the Safety Analysis Report remain valid.

The primary change to the Technical Specifications would be the LSSS. This was discussed in Section 2 above.

3.15 Financial Qualifications

The facility owner must demonstrate in their financial qualifications their ability to perform reactor operations in terms of staff compensation, radiation protection program commitments and ultimately the decommissioning of the facility. Considerations are made for cost of living adjustments as well as inflation.

There are no changes to the Financial Qualifications for this work.

4. OTHER SIMILAR TRTR UPRATES

4.1 Uprate Motivations

The power uprate of the PUR-1 reactor leads to a higher neutron flux within the fueled region of the core as well as in the experimental ports and irradiation facilities. The higher flux levels produce a facility more attractive to potential collaborative researchers. Material qualification, isotope production, and radiation resiliency are all highly reliant on a tailored radioactive field which is usually desired to be maximized.

4.2 University of Utah[12]

In June of 2009, the University of Utah filed to uprate power of its TRIGA reactor from 100 kW to 250 kW of steady state power. TRIGA reactors have pulse capability and allow for power transients but are often licensed for a steady state power level as well. The power uprate proposed remaining with a natural convective coolant, similar to the work described here. The reactor tank for the UUTR has 8000 gallons of coolant compared to the 6400 gallons available at the PUR-1 and a depth of 22 feet, compared to that of 17 feet at the PUR-1.

In Section 1.3.6 of the referenced Safety Analysis Report, the UUTR staff note the insufficient cooling capability of the UUTR heat exchanger to maintain steady coolant temperature below $16°C$ when the reactor is operated above 25 kW. Historic measurements of the pool temperature rise rate are $3°C/hour$ when the core is operated at 100 kW and were expected to be $8°C/hour$ when operated at 250 kW. They proposed an administrative control to limit the water temperature to $40°C$ during operations.

The University of Utah stated their purpose for limiting the reactor coolant temperature was to prevent skin scalding and limit degradation of the exchange bed resin. The physical hazard of limiting the water temperature to prevent inadvertent burns is not considered in this work but is another potential administrative limit which could be imposed.

The final proposed power for the University of Utah was chosen to be a static power level and not dependent on the radiation levels or the coolant temperature.

4.3 NC State[13]

In February of 2017, NC State filed a License Amendment Request with the US NRC to upgrade their reactor power from 1.0 MW to 2.6 MW. The school operates the PULSTAR reactor which has cooling capacity for up to 2 MW after a facility upgrade in 2013. The reactor pool has a capacity of 15,000 gallons of water. There is additionally a large biological shield with high density barytes to reduce radiation dose.

The NC State reactor features operational modes for both natural convection operation as well as forced convection flow. In the forced convective flow regime, they show a forced convection core flow rate vs. reactor power level scheme. Given an input forced flow rate, the reactor is able to operate safely in a range of one to five megawatts, although the license is up to 2.6 MW. Their proposed technical specification is shown below.

While this work shows that there have been considerations for a variable power limit as a function of a secondary parameter, it falls short of offering a licensed power which varies. The PUR-1 considerations in this work differ in the limiting input parameter (temperature vs coolant flow rate) however both show that there are dynamic levels at which a reactor may operate safely.

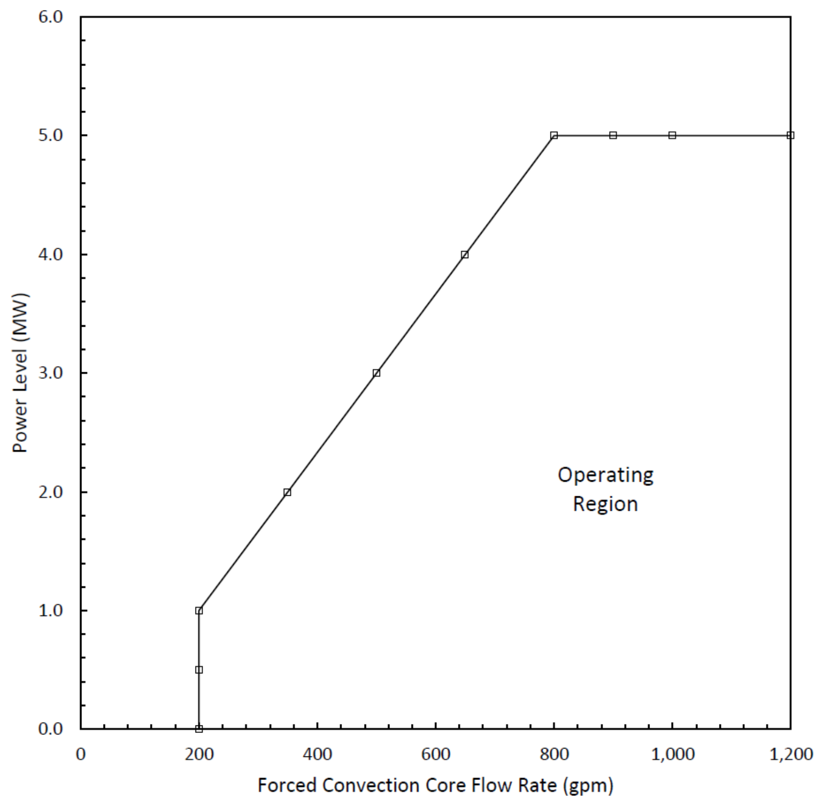

Figure 2: Forced convective flow throw the NC State PULSTAR reactor and corresponding power level. Taken directly from, Figure 4-15 and TS 2-1[13]

The figure above shows that with a forced convection core flow rate of 200 gallons per minute, the NC State reactor would operate up to 1 MW. It then extends this trend linearly from 200 to 800 gallons per minute offering power levels up to 5 MW. Above 800 gallons per minute, the reactor would be limited to 5 MW.

When operating in the natural convection flow mode, the NC State reactor is able to operate up to 1.0 MW although only operated in this mode at 250 kW. They restrict coolant temperature to $117\ °F$.

4.4 Other Power Uprates

Other power uprates have been performed at facilities such as Texas A&M[14], Reed College [15], Kansas State [16] and others. Most of these power uprates included investment into the respective facility for enhanced cooling capabilities, flow regulation, airborne effluent monitoring, or biological shield improvements. The reactors that have forced convection flow often include an allowance to operate the reactor at a lesser limiting power when the forced flow is not being utilized.

4.5 Summary of Selected Facilities

These two facilities show two important points upon which this work relies. The first is that at a research reactor, it has been proved permissible to operate the reactor at a power which exceeds the cooling capacity. Although the current PUR-1 license and the University of Utah cooling capacities were below their maximum power level, the NRC agreed that for reduced durations of full power operation, the facilities could be safely operated. Given the small core size and the volume of the pool, temperature rise rates should not exceed $10°/C$ per hour. At this rate of temperature rise, the operator or control system have ample time to respond to the elevated temperature and lower the reactor power accordingly.

Secondly, the NC State work shows that there are other facilities who have or may consider implementing a dynamic power operations portfolio. While they ultimately chose to have a single maximum power level, the referenced figure shows that this is not necessary. Provided they meet other safety, radiological, and occupational health limitations, there is room for continued growth in the maximum power.

5. SELECTED LIMITATONS ON POWER UPRATE

The purpose of this work is to investigate the power uprate capabilities of the PUR-1 reactor without significant investment in infrastructure. As discussed above, the facility has a long history of operations at low power, however, this limits the usefulness as a radiation facility and restricts the physical phenomenon which can be demonstrated in a teaching environment. The restriction on significant investment stems from the constrained operating budget present at most research reactors. While the PUR-1 has experienced a recent availability of funds due to the extreme growth of the College of Engineering, this one time influx will not sustain the reactor over a long period of time. Without a large private or public investment, the facility must operate within its current design.

5.1 Subcooled Light Water

The reactor is cooled by light water without phase transition. As the reactor power begins to increase, the bulk coolant volume has a corresponding rise in water temperature. At the interface between the clad and the coolant, nucleate boiling begins at higher power. Once phase transitions are considered, significantly different physics come into effect which are outside the scope of this work. The upgraded PUR-1 core will be limited to conditions where the reactor will stay below the onset of nucleate boiling (ONB) temperature.

The core is slightly undermoderated. [17] Therefore, if the onset of nucleate boiling power level were to be accidentally reached, the formation of voids within the coolant would produce a negative reactivity feedback and contribute to reactor power reduction.

5.2 Natural Convection

Forced cooling is another limiting factor for the PUR-1 core. Currently, all coolant flow is generated through natural convection. As the coolant heats during passage through the core, its density is decreased (due to temperature rise) and it is replaced by lower temperature coolant at the reactor inlet. The addition of forced cooling would increase the accessible power level significantly however this is not practicable with the current assembly and grid plate design.

5.3 10 kW Heat Removal Capacity

The reactor currently features 10.55 kW of heat removal capacity through the use of a shell and tube type heat exchanger. This unit is from the original construction of the reactor and is sufficient for current licensed operations. However, reactor power levels beyond this capacity could be permitted with the knowledge that the bulk pool temperature will rise as well. The 10 kW of heat removal would serve to inhibit temperature rise. The heat exchanger is utilized in this work but only to its rated value.

5.4 No Additional Shielding

One of the principle means of shielding operators and members of the public from heightened radiation levels is the reactor’s coolant (light water) in conjunction with the 18-inch concrete wall holding the coolant. The radiation levels at the top of the tank are generated from shine, airborne activated constituents, and any experiments which have been removed from the irradiation ports. Shine is the direct passage of radiation from the core, through the pool volume, and incident at the point of measurement. Much of this radiation is attenuated as it travels through the water, however, build-up effects keep the radiation levels at a non-negligible level.

The reactor was constructed nearly 60 years ago and there are concerns that any alteration of the tank may result in permanent damage. This would require a complete replacement and would likely be cost prohibitive for the facility’s lifetime budget leading to a permanent shutdown. Additionally, adding coolant or other shielding to the top of the pool is impracticable. The addition of coolant would require a tank extension. However, many of the control components (rod drive magnets for example) are currently located just above the water surface. A redesign of the control rod mechanism would be required and likely prohibitively expensive. Alternatively, the addition of a high-Z absorber is also not possible.

5.5 Core Configuration Unchanged

The models used for the estimation of the power levels utilize the standard PUR-1 reactor core. This standard 208 plate configuration is considered the upper threshold of loading with all plate locations filled. The PUR-1 license limits the excess reactivity of the core to $0.006\ \Delta k/k$. This limit is recognized in the modeled configuration. Addition of more reactor fuel could compromise this limit and necessitate the replacement or upgrade of control rods.

6. MODEL DETAILS

6.1 MCNP6

MCNP6 is a "general-purpose Monte Carlo radiation-transport code designed to track many particle types over broad ranges of energies." [18] The code allows the user to specify the geometry of the problem through the implementation of surface and cell cards. The bulk of the PUR-1 model relies on "macrobodies" or previously prepared arrangements of surfaces, which are then classified into cells. Each cell has a material identifier to reference its composition. Materials match either the vendor supplied isotopic information or are taken from a variety of sources. Particles are then generated as a point source at a specified location, volume source throughout a cell, or as a result of fission in fuel which was utilized for this model. The particles generated are then propagated throughout the model as pseudorandom numbers aid in determination of probabilistic collision, absorption and other physical events.

As the neutrons within the reactor are the primary particle of interest, the photons were neglected from this problem. To measure specific values, tallies are used in MCNP. A tally is the method of obtaining solutions to desired problems in the MCNP code. Usually the tally performs some sort of tracking of collisions, population, or track length of a given region. The two principle tallies used were the F4 and F7 tally an are discussed below. Note: MCNP also allows for the weighting of particles to improve computational performance and counting statistics. No weighting was used in this simulation.

6.1.1 The F4 Tally – Average Flux

The F4 tally measures the average value of the flux within a specified cell or set of cells. To calculate the value of the F4 tally, MCNP finds the location which a neutron enters the cell and its exit location. The total path length is then multiplied by the particles weight (unity in this problem) and divided by the volume of the cell. This gives the number of neutrons per area which are the units of flux. Mathematically, the F4 tally gives

$$F4 = \frac{\int \int \int \phi\left(\vec{r}, E, \vec{\Omega}\right) \, d\Omega \ \mathrm{dE} \ \mathrm{dV}}{V} \qquad \text{Eq. 1}$$

where ϕ is the neutron flux as a function of position $\vec{r}$, energy E, and angle $\vec{\Omega}$. The flux is normalized to the volume V. The energy range over which the flux is measured can be selected. For example, to find the thermal flux, which will be proportional to power, a corresponding energy range of 0 to 0.1 eV could be selected.

6.1.2 The F7 Tally – Energy Deposition

The F7 tally measures the fission energy deposition in the cell. It is an extension of the F4 tally but extended with a multiplier. The F7 total value is

$$F7 = F4 \cdot \frac{\Sigma_f \cdot Q}{\rho} \qquad \text{Eq. 2}$$

where Σ_f is the total macroscopic fission cross section of the cell, Q is the energy released in a fission event, and ρ is the gram density. The units of the F7 tally are in $MeV/gram$.

6.2 NATCON

Many plate type research reactors have the ability to run on natural convection alone. The NATCON code was developed by Argonne National Laboratory to simulate plate-type reactors operating in this space. The code assumes the pool temperature to be at a constant average temperature. The flow of the coolant through the core is driven by thermal expansion of the coolant and the density change created. The buoyant force generated from the heating is depressed by the viscous force in the created flow.

6.2.1 Buoyant Force

The buoyant force created by the thermal driver is given as

$$F_B = (\bar{\rho}_c - \rho_{AMB})V_C g \qquad \text{Eq. 3}$$

where F_B is the buoyant force, $\bar{\rho}_c$ is the average density of the water column, ρ_{AMB} is the density of the coolant in the tank, V_c is the volume of the channel, and g is the acceleration due to gravity.[19]

6.2.2 Frictional Forces

The buoyant force is depressed by the frictional forces of the flow. The frictional forces are given by the code as

$$F_F = \frac{(\rho v)_{in}^2}{2g} A_c \left[\frac{1}{2\rho_{in}} + \sum_{i=1}^{n} \frac{f \Delta z_i}{\rho D_H} + \frac{1}{\rho_{out}}\right] \qquad \text{Eq. 4}$$

with ρ being the density at the respective inlet or outlet, v is the velocity of the coolant, g as the acceleration due to gravity, f is the friction factor, Δz as the change in height in the subsection, and D_H as the hydraulic diameter of the coolant channel.

By setting these two forces equal to each other, the velocity of the coolant can be found. This coolant velocity is then used in the determination of the temperature of the fuel, clad, and coolant.

6.2.3 Onset of Nucleate Boiling (ONB)

The onset of nucleate boiling is a factor of the temperature of the wall, the coolant saturation temperature, coolant pressure, and heat flux. The Bergles-Rohsenow correlation used to find the wall temperature for the ONB is

$$(q/A)_{ONB} = 15.60 p^{1.156} (T_w - T_s)^{2.30/p^{0.0234}} \qquad \text{Eq. 5}$$

where q/A is the heat flux in $Btu/hr \cdot ft^2$, p is the pressure, T_w is the wall temperature, and T_s is the saturation temperature.

The ONB Ratio (ONBR) is one of the important outputs of the NATCON code. The ONBR is the ratio of the heat flux which would initiate nucleate boiling to the heat flux at the current node location.

6.2.4 NATCON Input and Output

The input and output from the NATCON code is given in a simple text file. The user is required to supply the number of standard elements in the core, the number of control elements, the number of plates in each, the dimensions of the fuel plate’s clad and meat, the channel dimensions, and safety factors for entrance effects, the pool depth, bulk pool temperature, the radial power peaking factor, and the axial values of power production. These axial values are obtained from output of the MCNP file as well as the radial power peaking factor. This radial factor is the plate’s burn rate above the average for all plates. The input deck for a sample of the reactor simulation is included as Appendix A.

The output of the NATCON code includes the heat flux, fuel centerline temperature, clad temperature, water temperature, water density, water pressure, saturation temperature, and the onset of nucleate boiling ratio. All of these are given as a function of channel height. The velocity of the coolant and the water temperature at the inlet and outlet are also given.

If desired, the user may also utilize a search function to find the power at which the ONBR ratio is equal to unity. The NATCON will find on the ONBR ratio as a function of channel height and determine the minimum value of the ratio. If this value is greater than one, the indication is the reactor has not achieved the onset of nucleate boiling. NATCON then reiterates the problem and increases the power of the core by $100\ kW$. If the ONBR is less than unity, the power is reduced until the ONBR is exactly one.

7. SUBSYSTEM POWER ENVELOPE ANALYSIS

7.1 Overview

This section will contain the principle results from a power uprate in each of several respective areas. Considerations include the neutronics calculations which give insight to the power distribution in the core, the heat removal capacity by natural convection of the bulk coolant volume, and radiation dose rate resulting from operation at higher power from both effluents and direct shine.

All data is simulated and there are no physical measurements. As the reactor's core was completely replaced in 2007 as part of the Global Threat Reduction Initiative, facility archived measurements are of a core with slightly different geometry (approximately 20 fewer fuel plates) and significantly different enrichment (93% compared to the current 20% enrichment). Due to chronic facility downtime and instrumentation maintenance issues, there are no known measured flux maps to date. These measurements are left as future work.

7.2 Neutronics

The PUR-1 core was modeled using MCNP6.2. The input deck used was originally developed in support of the 2007 fuel conversion and was subsequently modified with some corrections and to allow processing of desired data for this work. The core is a full model including the aluminum support grid, control rods, and approximate PUR-1 fuel loading.

7.2.1 Visualization of Transverse Flux Distribution

Several simulated measurements are needed to perform an accurate determination of limiting power factors with respect to neutronics. The first is the distribution of flux within the core. Figure 2 and Figure 3 below show the flux as a function of axial height and the transverse flux. These were created using an F4 tally implemented in a mesh from the neutronics code known as an "FMESH". The FMESH is a shorthand way to create a large number of F4 tallies without the need to individually code each one individually.

Figure 2 below shows the neutron flux of the core with an overlay of the cell lines at the centerline of the reactor. Each black line represents a barrier between different materials. The outer assemblies in the core are composed of graphite and serve as a neutron reflector while the inner assemblies contain the reactor fuel. The three assemblies which have plates in an alternate direction are the control assembles. SS1 is on the top right, SS2 on the top left, and the Regulating Rod on the bottom left.

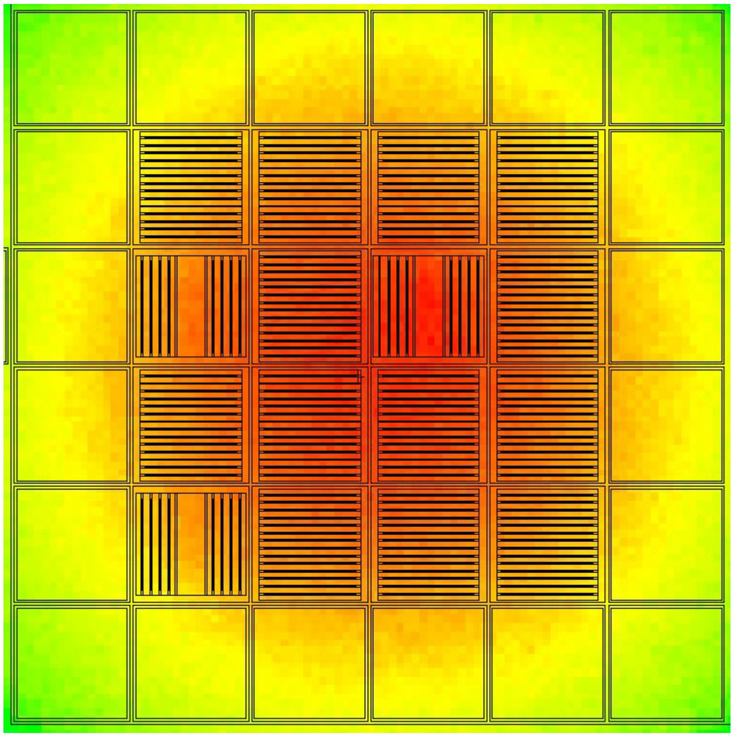

Figure 3: Representative thermal neutron flux heat map of PUR-1 core at 30 cm height, the reactor midplane.

It is clear here that the neutron flux is significantly greater in the central region of the core and there are some places within the loading where there is little to no contribution to overall power. Additionally of note in this figure is the variable worth of the graphite reflector surrounding the fuel. Many small reactors such as the PUR-1 require graphite due to the surface to volume ratio. These reflectors mitigate the neutron leakage. Note that the reflector elements on the corners are virtually unused and do little to promote reactivity.

A second artifact of the above flux heat map is the variable worth of the control elements. The first shim-safety, internally recognized as SS1, is the rod location to the top and right. Because of its more central position in the flux, its worth is much greater than that of the secondary shim safety which is located to the top left. Given two identical rods placed in these separate locations, SS1 would have a much larger impact to the neutron population. Finally, the third rod is located furthest from the peak flux and has the lowest worth. In the PUR-1 core, the third rod (lowest on the plot) is composed simply of stainless steel and has only slight amounts of boron contaminants found in all steel. Its principle means of control is through moderator displacement.

There is no analysis here of the modeling efficiency or the optimal mesh size to be run. The simulation used in this work is sufficiently simple that optimization of mesh size is not required. Multiple iterations were utilized to produce error bars for all relevant tally values of nominal magnitude.

7.2.2 Axial Flux Distribution

The axial flux distribution was found using a second FMESH tally. While the transverse flux was found using a two dimensional array of one centimeter cells, the axial flux is simply seeking the shape of the curve for thermal hydraulic analysis. Therefore, a one dimensional line of tallies was created which spans the height of the core and is located at the central channel between the four inner assemblies. Figure 3 below shows the thermal flux as a function of the height through the core and is given as a ratio of measurement to the peak value.

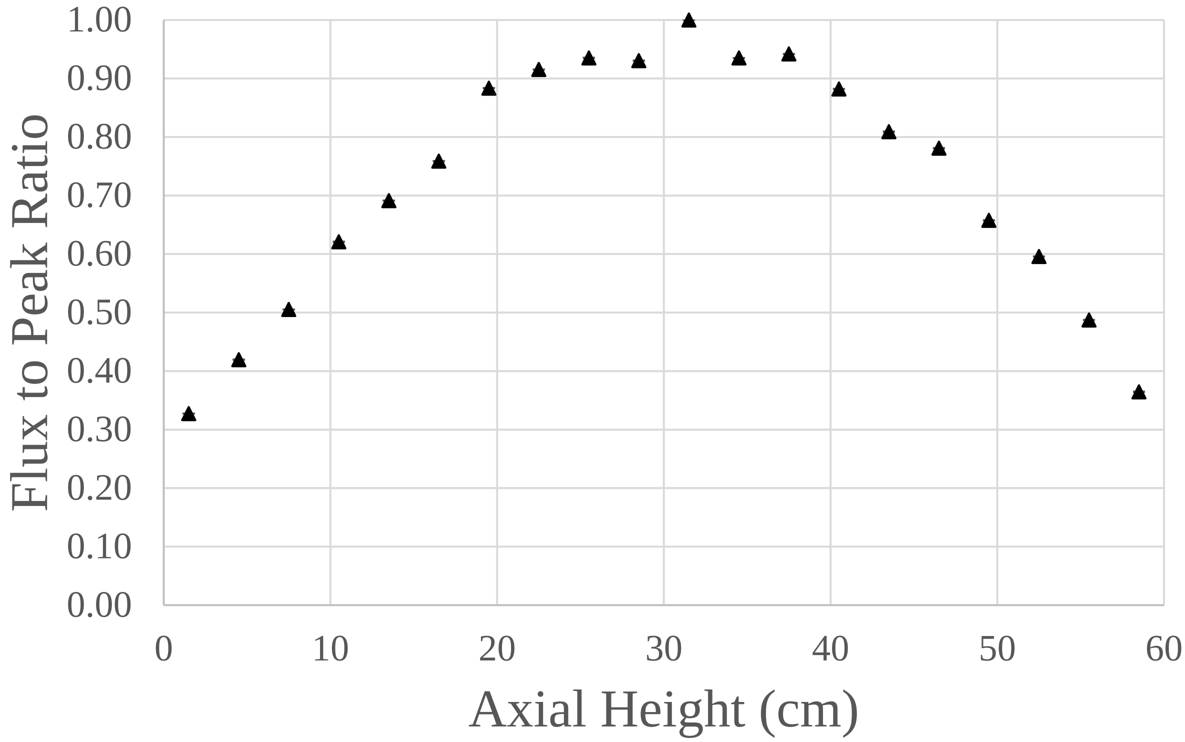

Figure 4: Flux to peak ratio of as a function of axial height in core

The upper and lower portions of the core experience approximately 33% of the flux at the maximum.

The true value for the flux is clearly directly related to the power level. Following the power uprate of the PUR-1 core from 1 kW to 10 kW, there have to date been no direct measurements of the flux. However, historically, the average thermal flux in the fuel region was determined to be $2.1 \times 10^{10}\ n/cm^2 s$ for operations at 1 kW.[17] For operations of 100 kW, this flux would then be extrapolated to be $2.1 \times 10^{12}\ n/cm^2 s$.

7.2.3 Maximally Burned Fuel Plate

In order to determine the plate with the highest power production within the core, an F7 tally was implemented. The F7 tally gives the fission energy deposition averaged over a given cell within the model. These tallies were utilized on all of the fuel plates of the fully loaded 208 plate core. The result of the F7 tally gives the total energy in MeV per gram of the cell per source particle of the simulation. Therefore, this result is simply multiplied by the mass of a cell and divided by the total from all the F7 tallies and the total mass of the core fuel. This gives the percentage of power produced in the particular cell (fuel plate). Regulatory guides limit the burn of the fuel in any one plate to 50% burn-up. Additionally, the manufacturer limits specifies the maximum recommended temperature of the fuel cladding be 530 C. This temperature limit is further discussed in the thermal hydraulic conclusion section.

The highest burned plate contains 0.852 ± 0.002 % of the total power in the core and will be a limiting factor in total core power should this plate's temperature begin to approach upper limits. There are four fuel plates which receive over 0.75% of the core's power each and eight which are above 0.7%. The top 10 plates collectively carry 7.5% of the total core power. Figure 4 below shows the distribution of power produced in particular plates for the core using this model. Each plate's fission energy deposition is found and normalized. Following normalization, the plates are sorted from the highest burn to the lowest and plotted.

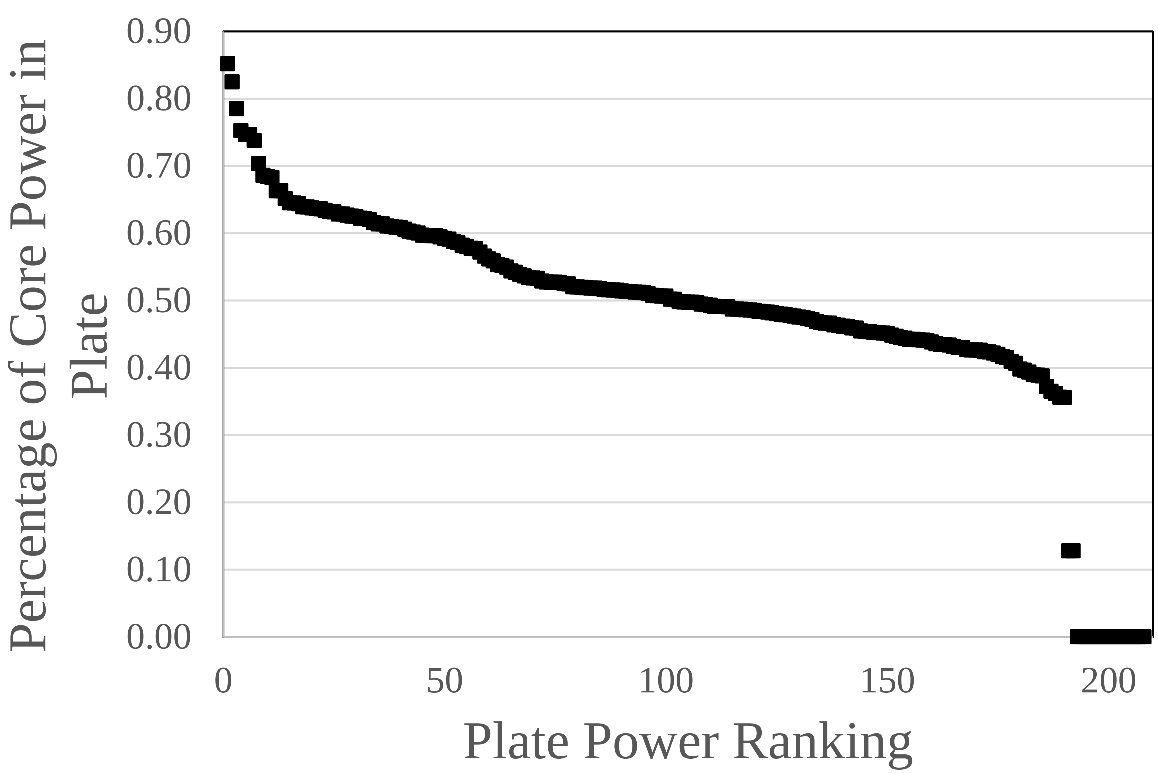

Figure 5: Ranked plate power by percentage of total core power

There are some important and interesting features of this plot. As discussed above, the first 10 plates carry 7.5% of the total core power. There are several plateaus in the plot which give information on the loading of the core. The flat region between ranked plates 15 and 50 contain those plates which are principally in the central four assemblies. With each standard assembly having 14 plates and the control assemblies with eight plates, this gives 50 in the central region. The second plateau from plates 50 to 190 shows a near constant utilization. Finally, this plot also shows some plates in the core represent such small power production their worth is negligible and they are unneeded for operation. There are 18 of these near zero power plates.

7.3 Thermal Hydraulic Analysis

7.3.1 Model Assumptions

The NATCON code takes the bulk pool temperature to be constant as a function of time. This mixing model is of sufficient accuracy for a first order model as the coolant flowrate through the core at the ONB power is at least 3.5 cm/s. This would indicate the total coolant volume would circulate through the core over a minimum of 72 minutes.

All temperature limitations, maximum coolant velocities, and onset of nucleate boiling values are given for the hottest burned plate in the core. This gives the most conservative values for the limitations on the core power. The mass flow rate from the model gives the total core mass flow rate.

7.3.2 Onset of Nucleate Boiling Over Temperature Ranges

The Onset of Nucleate Boiling is a limitation not surpassed in this work as discussed above. The NATCON code features a search feature to find the ONB power level. It is clear the power level which will induce boiling is a function of the original coolant temperature entering the core. Therefore, as the bulk pool temperature rises, the ONB power level will decrease accordingly. Figure 5 below shows the ONB power level as a function of the bulk coolant temperature.

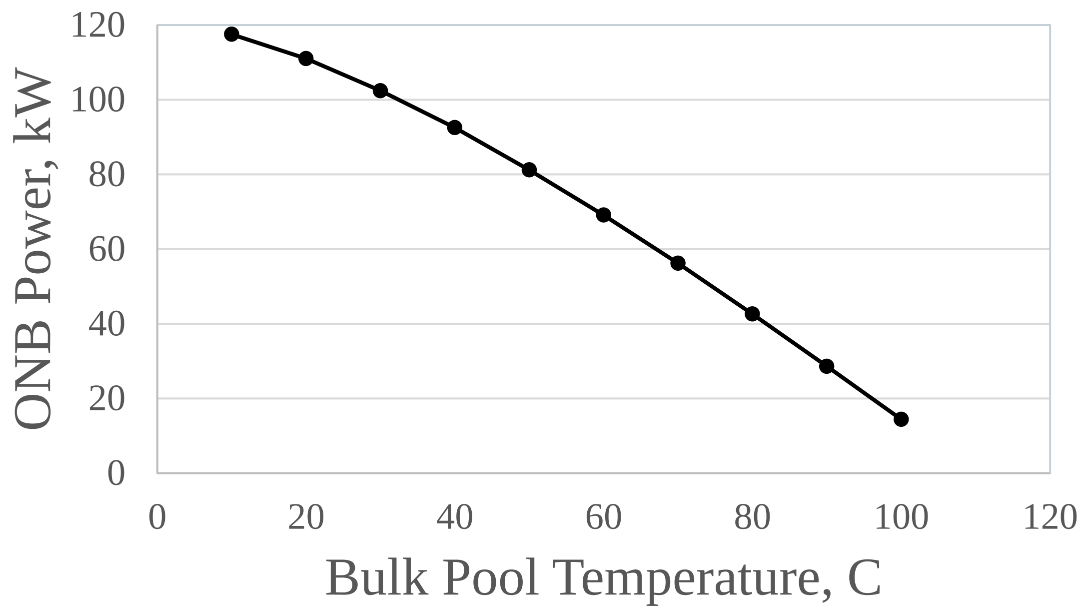

Figure 6: Bulk pool temperature as a function of ONB Power

The ONB Power level is not zero at $100\ °C$ for two reasons. Firstly, the reactor will reject some heat from the surface of the pool and to the surrounding support structure. Secondly, the boiling temperature of the water will be increased as the pressure is included underneath the water column. Finally, the function of the heat exchanger will remove at least $10\ kW$ of heat.

Here, it is clear that in order to achieve a substantive increase in the reactor's power while maintaining a threshold below the onset of nucleate boiling, the reactor's operating time would be limited to maintain a sufficiently low bulk coolant temperature.

7.3.3 Temperature Rise of Coolant Through Core

As the coolant enters the core at a temperature equal to the bulk pool temperature, it comes into contact with the fuel plates. The fuel plate temperature is a function of the number of fission events and therefore the thermal flux shown above. The coolant absorbs this thermal energy to maintain the cladding temperatures at safe levels.

The NATCON code provides the fuel, clad, and coolant temperature as a function of axial height in natural convection cooled plate type reactors. Figure 6 below shows these temperatures along the height of the core at a bulk pool temperature of $30\ °C$ and a reactor power level of $102.4\ kW$. This power level is the predicted ONB power for a bulk pool temperature of $30°C$. With this inlet temperature and reactor power level, the onset of nucleate boiling ratio is unity at 34.5 centimeters, or 4.5 centimeters above centerline.

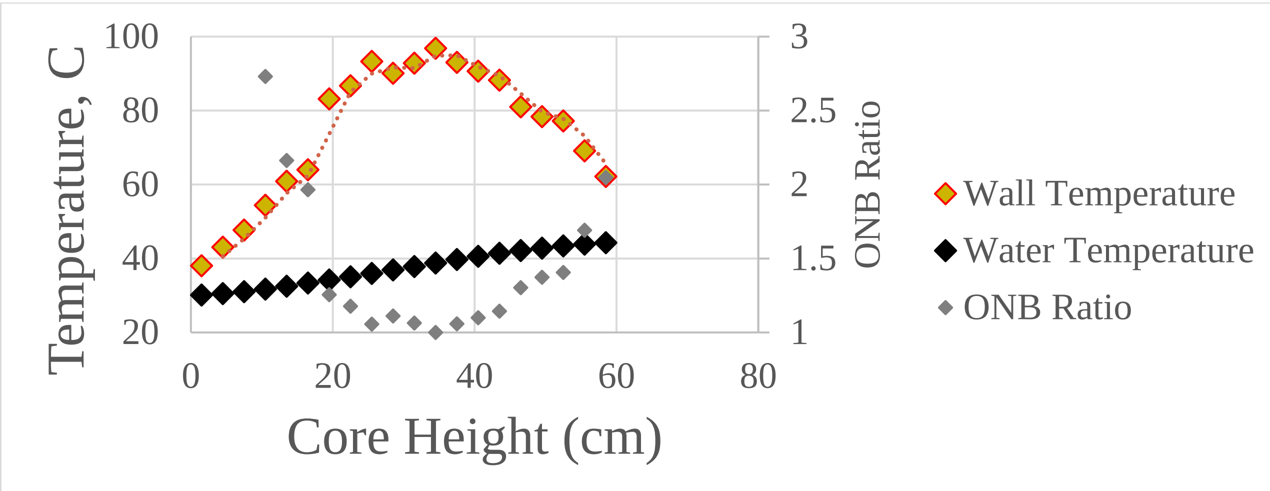

Figure 7: Fuel and coolant temperature along axial height of core with ONB Ratio for the hottest plate

For clarity, the interface between the clad and the coolant is the location of the onset of nucleate boiling. The subcooled water, having a temperature between $30°C$ and $45°C$, begins to nucleate due to the high wall heat flux.

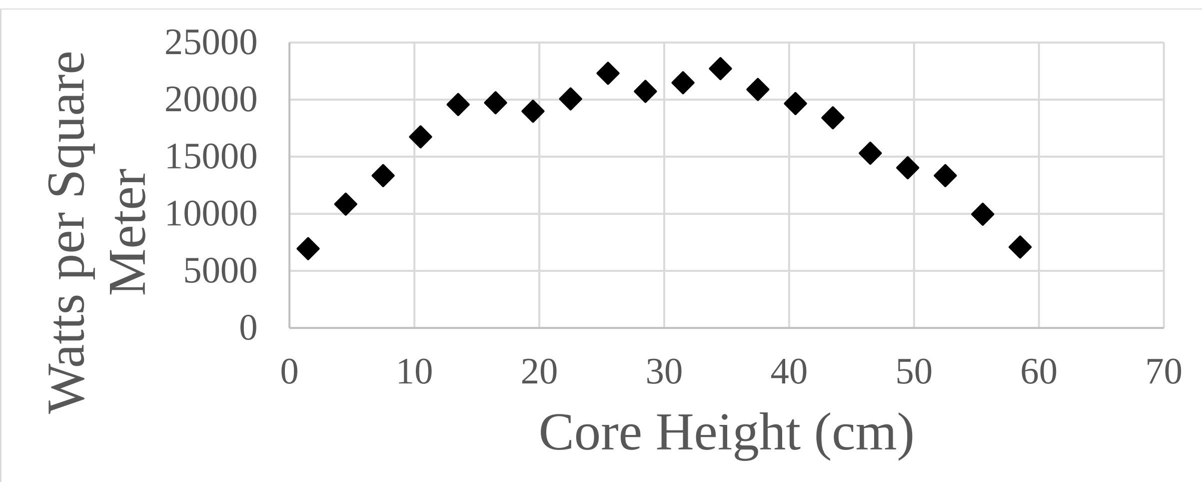

Figure 8: Hot Fuel Plate Wall Heat Flux

7.3.4 Pool Heat Rate

The rate of pool temperature heat rise is discussed in the PUR-1 Safety Analysis Report. The change in temperature as a function of heat capacity, mass and heat addition is

$$\Delta T = \frac{Q - Q_{out}}{mc} \qquad \text{Eq. 6}$$

where Q is the generation of heat, Q_{out} is the removal of heat by the heat exchanger, m is the mass of the water, and c is the heat capacity of the water. The generation of heat is the product of the power P and time t

$$\Delta T = \frac{(P - 10.55\ kW)t}{mc} \qquad \text{Eq. 7}$$

The heat rate of the pool can then be plotted as a function of the bulk pool temperature when the reactor is operated at the ONB Power level. That is, if the reactor's coolant temperature were at the specified value and the core's power was taken directly to the level whereby NATCON would predict the onset of nucleate boiling to begin, the y-axis gives the rate of temperature rise per hour.

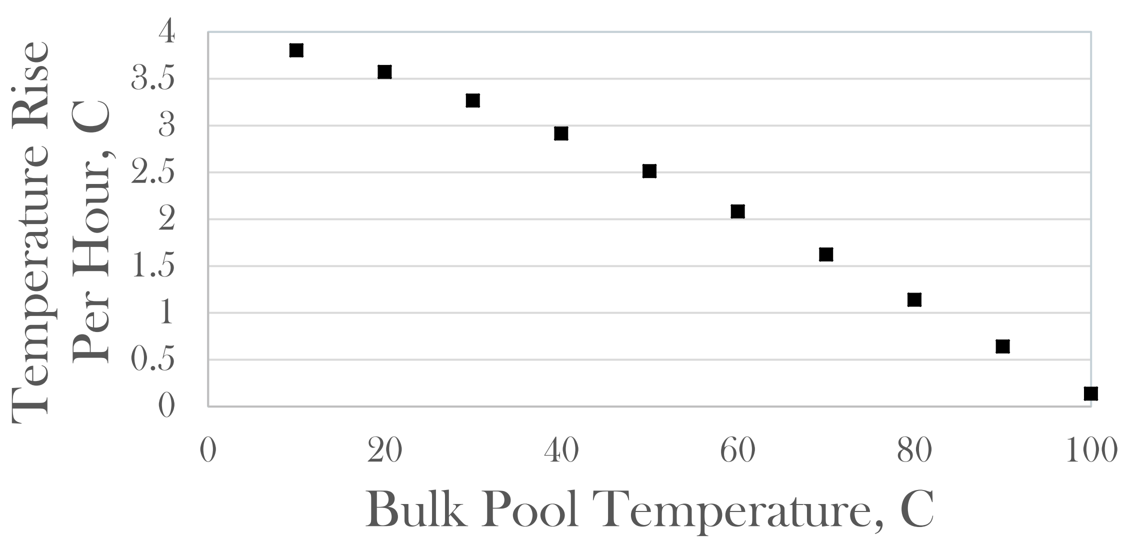

Figure 9: Temperature rise of bulk pool volume as a function of the reactor's coolant temperature and operations at ONB Power.

An example is given to aid in reading this important figure. If the reactor's bulk coolant temperature were at $40°C$, the coolant would experience the Onset of Nucleate Boiling at $92.5\ kW$, a value taken from Figure 5. With the reactor operating at $92.5\ kW$ and the heat exchanger running at full capacity, the bulk coolant volume would heat by $2.9\ °C/hour$. Therefore, this figure shows that a power uprate will not only be limited by the inlet bulk pool temperature but also by the length of time of operation. In order for operations to occur without nucleate boiling, the reactor's power must be continually decreased to maintain a level below this curve.

7.4 Airborne Effluent

As atoms of natural air are dissolved in the water, those atoms can be activated as they pass by or through the core volume. They can then be liberated from the pool as the water rises to the pool surface. These airborne activated atoms can then be inhaled by facility staff, or eventually members of the public. Dose limitations are placed to prevent undue exposure from the source.

A principle value in the determination of the dose from airborne effluent is the mass flow rate through the core. The NATCON code was used to determine this mass flow rate as a function of the pool temperature at the ONB power level. Again, the listed values here assume the bulk pool temperature shown on the x-axis and a reactor power level whereby the onset of nucleate boiling is just beginning.

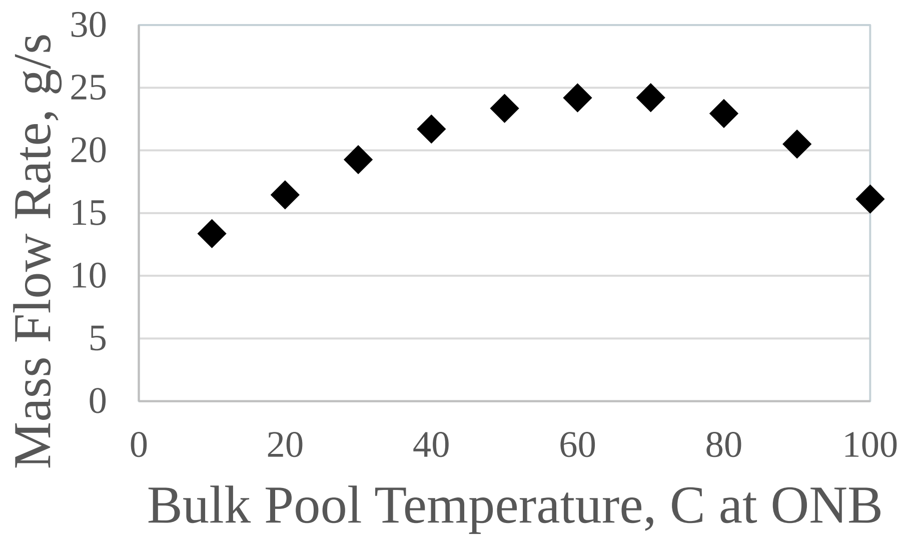

Figure 10: Average mass flow rate through channels given bulk pool temperature and operations at ONB Power level.

The mass flow rate is dependent on the maximum change in temperature as the coolant passes through the core. At high temperatures, the mass flow rate is diminished due to the lack of thermal difference between the plates and the coolant. Therefore, there is less change as the coolant passes through the core and less driving head.

7.4.1 Argon-41

One of the primary radiological concerns in pool type reactor operation over long periods of time is the buildup of radioactive argon within the reactor bay. In the Earth's atmosphere, roughly 1% of the air is composed of argon-40. As the air is absorbed in the water, these atoms will pass through the core with the coolant, have the possibility of abosorbing a neutron, and become radioactive argon-41.

$$^{40}Ar + n \rightarrow {}^{41}Ar + \gamma \qquad \text{Eq. 8}$$

That coolant carrying the radioactive Argon-41 rises to the pool surface where the argon can exchange with room air. Although the the argon-41 decays as it moves through the coolant and within the reactor room, the level of argon radioactivity can rise to unacceptable levels causing health concerns.

The following determination of the Argon dose rate in the reactor room directly follows the method outlined in prior work from Section 11.1.b.ii in the Safety Analysis Report.[11] This is extended for the variable flow rate through the core at different coolant flow rates.

The dose rate for a nuclear worker within the reactor bay is found by determining the activity of Argon-41 released into the room atmosphere, immediately and completely dispersed into throughout the available space and continuously removed through the reactor's exhaust fan. With an atmospheric pressure of $101{,}325$ pascals, the argon partial pressure is $1{,}013$ Pa. Henry's Law gives the atom density of gasses that are dissolved into a volume of fluid. For natural argon, this number is $1.4 \times 10^{-5}\ mol/m^3Pa$. Using this constant and the partial pressure of the argon, the atom density in the water is found to be $8.54 \times 10^{15}\ atoms/cm^3$. This is then corrected for the 99% of all argon which is ^{40}Ar. The mass flow rates taken from reactor operations at the ONB power level then give the number of argon atoms flowing through the core assuming a water density of 1 $gram/cm^3$.

The time for total recirculation of the pool must be at least

$$T_{circ} = \frac{V_{pool}}{\dot{V}_{core}} \qquad \text{Eq. 9}$$

where T_{circ} is the time it takes for the water to circulate through the bulk coolant volume, V_{pool} is the volume of the pool, and $\dot{V}_{core}$ is the volumetric flow rate through the core. With the volumetric flowrate from NATCON, the time for circulation ranges from one hour to four. The activity of the argon atoms during this flow period is found from the number of atoms, the thermal cross section of the argon, the reactor flux and the time over which the atoms have to decay while approaching saturation.

$$A_{sat} = \frac{N\sigma_{th}\left(1 - e^{-\lambda t}\right)}{1 - e^{-\lambda(t+T_{circ})}} \qquad \text{Eq. 10}$$

where t is the time for the atoms to transverse the core, N is the atom number density, σ_{th} is the thermal neutron cross section, and λ is the decay constant for the argon. The argon gas which is within the pool volume then exchanges with the reactor room air. This exchange S is modeled as

$$S = 0.93BN_{Ar}A_{surf} \qquad \text{Eq. 11}$$

where B is the surface exchange coefficient and A_{surf} is the exposed surface area of the pool. Similar to the core, the activated Argon will take time to saturate the reactor room volume. This time is expanded as the exhaust fan continues to expel air from the reactor bay. In a manner similar to the pool saturation rate, the reactor room effective half-life for airborne Argon is $35.3\ minutes$ as taken directly from the PUR-1 SAR. Dose conversion factors are available which give the dose rate in $mRem/hour$ per $\mu Ci/cm^3$. For Argon this is $8.03 \times 10^5\ (mRem/hour)/(\mu Ci/cm^3)$. By determining the activity per volume within the reactor room, the dose rate is calculated.

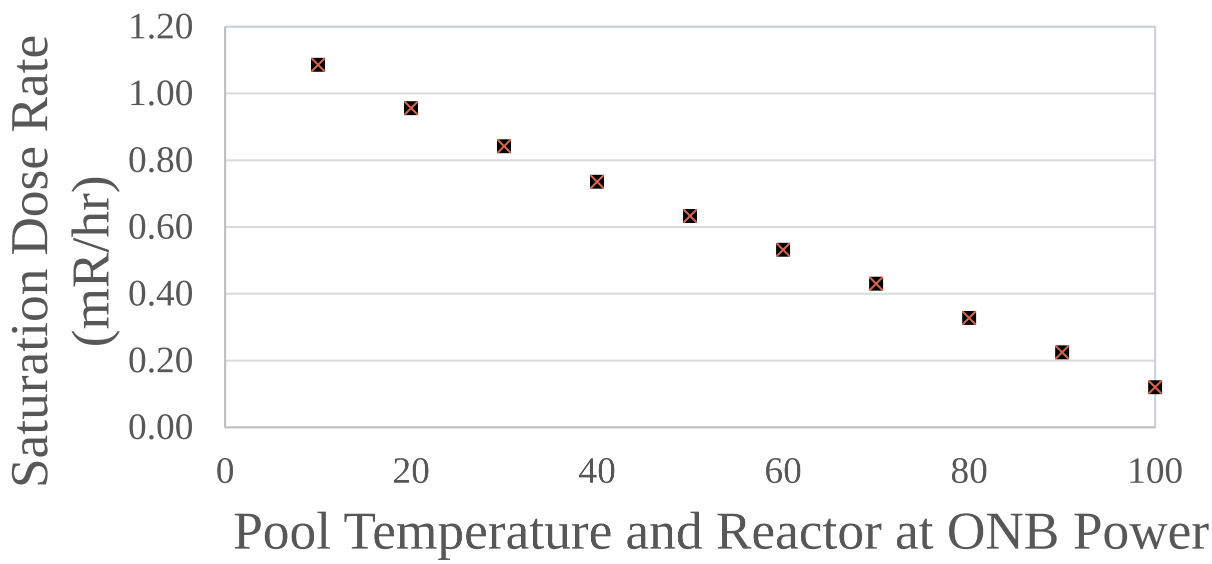

Figure 11: Reactor room dose rate given operations at specific pool temperatures and ONB power level

This plot shows the saturation dose rate is the highest for steady state operations at lower pool temperatures. The effect stems from the higher available power for lower pool temperatures. Consider two power levels: $120\ kW$ with the coolant at $10°C$ and $69\ kW$ with the coolant at $60°C$. Considering the higher power level, the coolant flow rate is 60% of that at 69 kW. However the flux is nearly double at the higher power level. With greater flux, gives more argon activation and therefore higher dose rate.

7.4.2 Nitrogen-16

Radioactive nitrogen is a second concern at some pool type research reactors, especially those with forced cooling capacity. Earth's atmosphere is approximately 78% nitrogen of which 99.6% is nitrogen-14 and 0.4% is nitrogen-15. As these isotopes pass through the core in the same manner as the argon, they can become activated to be nitrogen-16 or nitrogen-17. These have half-lives of 7.1 and 4.2 seconds respectively. If there is sufficient coolant velocity, stable upward flow, and large enough flux, buildup of the radioactive Nitrogen can cause radiological concerns as well.

Figure 10 below shows the coolant velocity through the core as a function of operations at given bulk pool temperature and ONB Power.

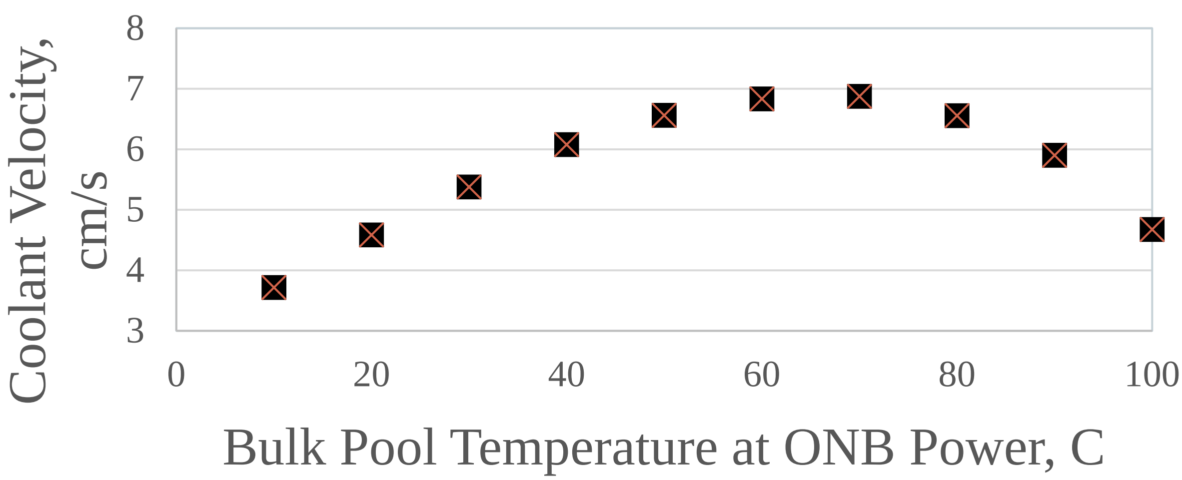

Figure 12: Coolant velocity through core at given temperature and ONB Power

The maximum velocity is 6.8 cm/s. Recall the depth from the top of the pool to the top of the reactor core is 13 $feet$ or 396 cm. Supposing a perfectly upward flow with no mixing, it would take 58 $seconds$ for the coolant to reach the pool surface. With a half-life of 7.1 $seconds$, this represents over eight half-lives. 99.7% of the nitrogen has decayed and the nitrogen activation is therefore a non-factor.

7.5 Core Shine

The top of the reactor core is 13 feet below the surface of the reactor pool. This 13 feet of water is the principle means of protection against direct radiation exposure during normal operations. The dose rate of a photon flux is given by

$$\dot{\mathcal{D}} = \phi(\vec{r}) k_D B_D(E_0, b, Z) \tag{Eq. 12}$$

as stated by Wallace [20] where ϕ is the flux, k_D is a flux-to-dose conversion factor, B_D is the build-up as a function of photon energy E_0, material compositions b, and the equivalent atomic number Z while assuming the core is a point source. Wallace compiles the build-up factor coefficients used to calculate b for a variety of materials including water, the principle shield utilized in the PUR-1 core. The value for b is given by

$$b = \sum_{i}^{N} \mu_i t_i \tag{Eq. 13}$$

where μ_i is the macroscopic gamma-ray attenuation coefficient for material i and t is the thickness of the given material. Note the units of b are in mean-free-paths. The flux from the dose rate above will decrease as a function of $1/r^2$ as well as the attenuation of photons as they move through the media. Because the flux and the build-up are both functions of energy, the true dose rate should be integrated over energy as

$$\dot{\mathcal{D}} = \int_0^\infty \frac{S \cdot e^{-\mu r}}{4\pi r^2} k_D B(b, E_0, Z)\, dE \tag{Eq. 14}$$

This equation matches that as found in the PUR-1 SAR.

The Build-up Factor is reported in a variety of forms however the one reported by Taylor and commonly used is

$$B(\mu r, E) = A(E)e^{-\alpha_1(E)\cdot\mu R} + \big(1 - A(E)\big)e^{-\alpha_2(E)\cdot\mu R} \quad \text{Eq. 15}$$

where A, α_1, and α_2 are simply tabulated coefficients as compiled by Wallace. The emission of the photons from fission was reported by Peelle and Maienschein in [21]. Their results were simplified and are given in Table 1.

Table 1: Prompt photons released after thermally induced fission of U-235

Energy (MeV)	**Photons Per Fisson**
0.5	5.2
2	1.8
4	0.22
6	0.025
8	0.002

The ultimate dose rate at the top of the pool level is linear with respect to the source intensity or power level. The dose rate at the top of the pool is

$$\dot{\mathcal{D}} = 0.18\frac{mRem/hour}{kW} \quad \text{Eq. 16}$$

The plot below shows expected radiation levels at the top of the pool.

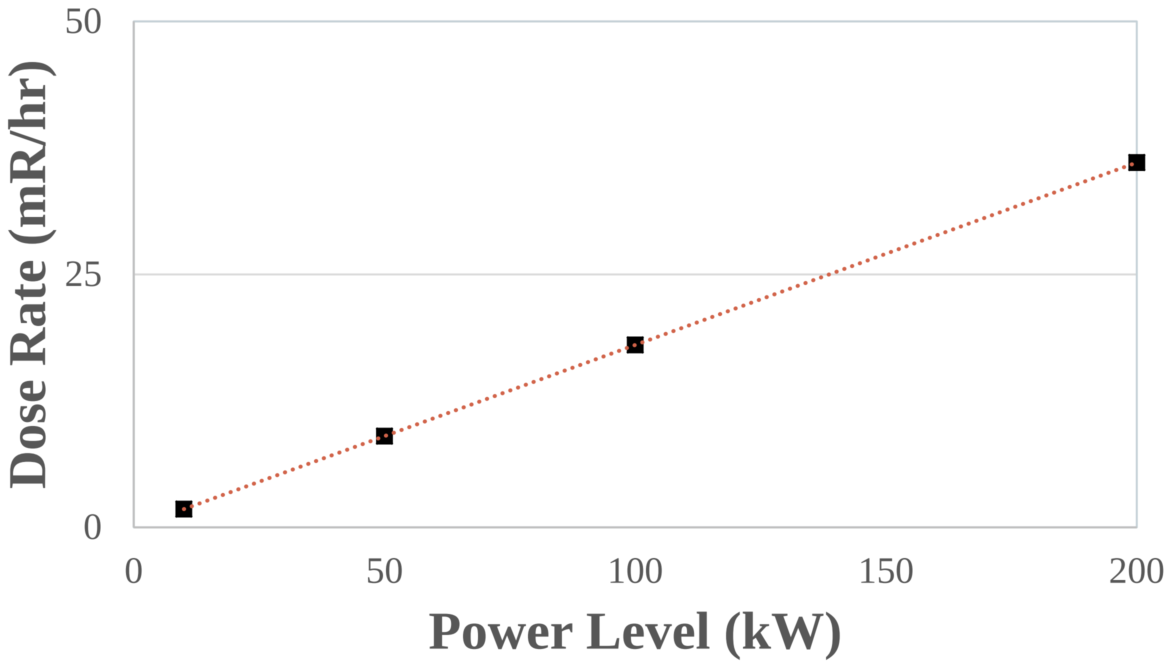

Figure 13: Pool top radiation level from shine as a function of power level.

8. SUMMARY OF LIMITING FACTORS

8.1 Plate Power Limitations

The peak burn in a single fuel plate was found to contain 0.852 ± 0.002 % of total core power. While the fuel can easily be replaced, it is instructive to understand the operational lifetime of the plate at elevated power levels. Assuming the reactor to operate at $200\ kW$ and $24\ hours/day$ for $365\ days/year$,

$$200 \times 10^3\ W \cdot \frac{1\ J/s}{1\ W} \cdot \frac{6.24 \times 10^{18}\ eV}{1\ J} \cdot \frac{1\ fission}{200 \times 10^6\ eV} \cdot \frac{1\ {}^{235}U\ atom}{1\ fission} \cdot \frac{235\ gram}{6.022 \times 10^{23}\ atoms} \cdot 0.00852 \frac{Plate\ Power}{Core\ Power} = \frac{Grams}{Second} \quad \text{Eq. 17}$$

$$2.075 \times 10^{-8} \frac{grams}{second} \cdot \frac{3600\ seconds}{1\ hour} \cdot \frac{24\ hours}{1\ day} \cdot \frac{365\ days}{1\ year} = 0.65\ grams\ {}^{235}U\ per\ year \quad \text{Eq. 18}$$

This gives a burn-up of 5.2% in one year for the hottest burned fuel plate. Of course, some plate and assembly shuffling could be done to extend the life of the plate however this operational threshold represents values never previously seen by the PUR-1 and is an upper bounding case. At this burn rate, even the hottest plate could be fully utilized over a decade with no fuel configuration changes.

8.2 Plate Temperature Limitations

The maximum temperature experienced by a plate which is restricted to heat fluxes which will not initiate nucleate boiling are far below those which are limited by the manufacturer. The PUR-1 Technical Specifications have a safety limit for the plate to maintain cladding temperatures below

$530°C$. In the example solution shown 6.3.3, the maximum fuel temperature was $112.5°C$. This is 20% of the limiting value and will not inhibit any higher power.

8.3 Pool Heat Rate

The pool heat rate will be a major limiting factor for the maximum achievable reactor power. As the bulk pool temperature continues to rise, the power level which will initiate the ONB continues to drop as shown in Figure 7 above. Operating the reactor at the ONB power level ($102.4\ kW$) with a bulk coolant temperature of $30°C$, gives a temperature rise of $3.27\ °C$ over the course of one hour. This temperature rise then limits the reactor power to $99.2\ kW$.

These two factors continue to play off each other and steady state operation will diminish the allowed reactor power over time to maintain a coolant which is not undergoing phase transition.

8.4 Airborne Radioactive Effluent

The radiation exposure to a worker in the reactor bay produces some nominal level of dose. However, this airborne calculation relied on the continuous operation of the facility over long periods of time. The previous section details how over the course of several hours, the reactor power is forced to decrease to compensate for the initiation of ONB. The higher the bulk pool volume's temperature became, the lower the saturation dose rate in the reactor bay. Additionally, the maximum dose rate was found to be $1.09\ mRem/hour$. This dose is sufficiently low to not inhibit work in the environment and is a non-factor in the operational power of the facility.

8.5 Reactor Shine

The shine from the core and the directly measured radiation levels associated create an administrative limitation on safe operation of the reactor. The NRC defines a radiation area as "Any area with radiation levels greater than 5 millirems in one hour at 30 centimeters from the source or from any surface through which the radiation penetrates." The dose rate at the top of the pool is given by Eq. 16.

Under this definition and the estimated dose rate, the reactor pool top becomes a radiation area at $27.7\ kW$. The NRC designates high radiation areas to have dose rates of $100\ mRem/hour$. This value would occur at a reactor power of $555\ kW$. The occupational restrictions placed by the Radiological and Environmental Management group would therefore have to come into consideration for reactor power.

With a coolant temperature of $10°C$, a limiting value which would require chilling rather than heating of the pool, the ONB power level found above was $117.5\ kW$. At this power level, the estimated dose rate is $21.2\ mRem/hour$. Assuming this dose rate to be acceptable for a radiation worker, the reactor shine is not a direct factor in limiting reactor power.

8.6 Safety Margin

Some consideration must be given to the uncertainty in the power level measurement performed by the neutron flux monitors. Thus far, a safety factor was not discussed. Historically, the reactor was licensed with a 50% instrumentation uncertainty. This number will be significantly reduced with the installation of new digital instrumentation and control system. Assuming their uncertainty to be less than 10%, a safety factor of 1.1 should be included to account for the uncertainty in power determination.

8.7 Future Work On Limitations

This thesis has outlined how a dynamic limiting power level could be chosen as a function of the radiation level and the bulk coolant temperature. The main thrust for future work to be done which will further prepare the reactor staff for a License Amendment Request (LAR) to the NRC would be to extend the analysis for the accident scenarios traditionally discussed in Chapter 13 of the SAR. These accident scenarios include the failure of an experiment, failure of a fueled experiment, fuel handling accident, insertion of maximum excess reactivity, and loss of coolant accident. At higher reactor powers, there will be a higher saturation activity during isotope production work which would then release more radioactivity into the reactor bay air during an accident. Additionally, with higher reactor power, the fuel plates would have more volatile fission products within recoil range of the exposed plate exterior. As these fission products are liberated in an accident scenario, their impact on the health and safety of workers and members of the public must be considered. Finally, with the reactor at a higher power, it would achieve greater maximum temperatures in a ramped or sudden reactivity insertion accident.

These additional considerations may provide further restraint on the reactor power which would be included, with a margin of safety, to the final operations power envelope.

9. CONCLUSIONS

The limiting reactor power level was investigated to determine the maximum power at which the reactor could safely be operated without a major change in facility design or administrative procedures. The most limiting factor was the onset of nucleate boiling power level. This is the power at which the coolant will start to nucleate given a certain bulk pool temperature. As the reactor is operated, the coolant continues to heat and the ONB power level diminishes. Therefore, in order to achieve the highest flux, the reactor would be limited to short operational periods before the bulk volume temperature is affected.

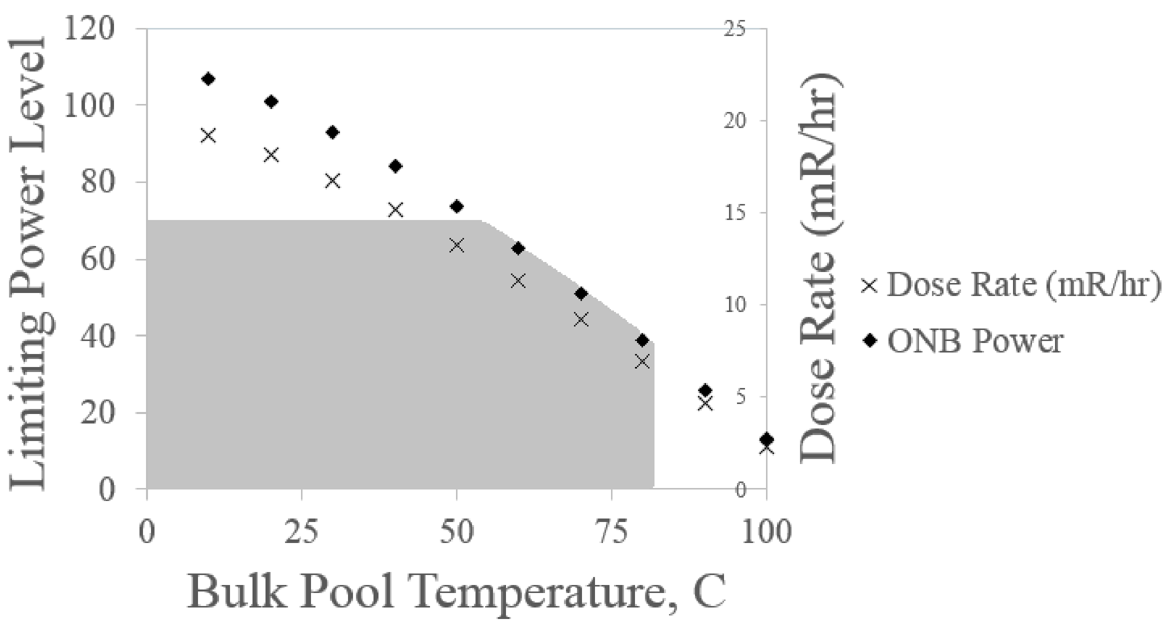

Figure 14: Example power level limitations on reactor operation included administrative limits for bulk pool temperature, dose rates, and staying below the ONB.

This final figure shows the reactor power limitations as a function of the bulk pool temperature and dose rate. This power level is reduced by a factor of 10% to account for instrumentation uncertainty and an administrative bulk pool temperature limit of $80°C$. The dose rate is included here to illustrate how an administrative limit may affect the limiting power level. Steady state operation must be performed under the envelope created by the ONB Power listing and the radiation dose rate limit. For example, operations at $60\ kW$ with a bulk pool temperature of $25°C$

would be permissible while operations at $80\ kW$ with a bulk pool temperature of $25°C$ would not due to dose rate. Additionally, operations at a bulk bool temperature of $75°C$ and a power level of $25\ kW$ would be permitted while operations at the same temperature at $60\ kW$ would not be permitted due to the Onset of Nucleate Boiling.

Consider another sample operation of the reactor where the facility has a researcher who is looking to irradiate a sample in a neutron flux of $1.25 \times 10^{12}\ n/cm^2s$. Given the reactor has a reported flux of $2.1 \times 10^{10}\ n/cm^2s/kW$, this operation would need to occur at 59.5 kW. During operations at this power level, the reactor would heat the bulk coolant volume at a rate of $2°C/hour$ as shown in Figure 9. The usual start-up temperature of the coolant volume is approximately $26°C$. When the reactor is operated at 59.5 kW, the coolant volume would have a maximum allowed temperature of approximately $58°C$ per Figure 14 above. This would therefore allow for 16 hours of continuous operation at this power level before the core power would have to be reduced to stay within the operational envelope. The dose rate would be $10.7\ mRem/hour$ and would not be a limiting factor in this scenario.

The work described considers no change to the facility. Some future work would enable longer durations (potentially indefinite) of operation at elevated power levels. A forced cooling system is impractical given the design of the core support structure and is not considered. Without the addition of a forced cooling system, the principle means of improving the operational capabilities of the reactor would be to maintain the coolant temperature at a sufficiently low level by expanding the heat exchanger capabilities. The current heat exchanger has a capacity of $10.55\ kW$. A higher capacity, up to the limit of the power availability without nucleate boiling for room temperature water, would maintain the reactor coolant temperature within the specified ONB envelope.

Finally, relocating the control console outside of the reactor bay would alleviate many of the radiological concerns to members of the staff from both direct core shine and airborne effluent. Adjacent space to the reactor room is available which could be retrofitted to house the console. Discounting facility staff time for movement and licensing changes, the expense of this relocation would be minimal.

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APPENDIX A

A sample input deck for the NATCON simulation is provided here for reference.

```
PUR-1 LEU
   20     1    13    14     3     8    0      0
0.600075    0.059563    0.000508    80.0        0.000381    180.0
0.637794    0.071933    0.005004    4.620514    0.01143     100.0
0.000001    0.02        1.6404      0
1.00        1.50        1.250       1.250       1.250       1.250
0.0         0.000
0.050       0.453
0.100       0.581
0.150       0.700
0.200       0.860
0.250       0.957
0.300       1.050
0.350       1.220
0.400       1.270
0.450       1.300
0.500       1.290
0.550       1.380
0.600       1.290
0.650       1.300
0.700       1.220
0.750       1.120
0.800       1.080
0.850       0.911
0.900       0.825
0.950       0.675
1.000       0.505
```