Title

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Thermal Hydraulic Calculations to Support Increase in Operating Power in McClellen Nuclear Radiation Center (MNRC) TRIGA Reactor*

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1.0 Introduction

The McClellen Nuclear Radiation Center (MNRC) reactor was previously licensed to operate at an operational safety limit of 1.1MW and was known as the Stationary Neutron Radiography System. The thermal and hydraulic design is described in detail in the Safety Analysis Report (References 1 and 2). A new analysis was required to show that the MNRC could be safely operated at powers in excess of the 1.1MW safety limit. The required analysis has been performed using the RELAP5/MOD3.1 computer program (Reference 3). The RELAP5 code was developed for the USNRC by the Idaho National Engineering Laboratory (INEL) to analyze transients and accidents in light water reactors. The RELAP5 code is highly generic and can be used to analyze a wide variety of hydraulic and thermal transients involving almost any user defined nuclear or non-nuclear system.

The MOD3 version of RELAP5 has been developed jointly by the NRC and a consortium of several countries and domestic organizations that are members of the International Code Assessment and Applications Program (ICAP). The RELAP5/MOD3 development program included many improvements based on the results of assessments against small-break LOCAs and operational transient test data.

A RELAP5 model consists of a system of control volumes which are connected by flow junctions. The fluid mass, momentum, and energy equations along with the appropriate equation of state are solved for the user defined geometry. The RELAP5/MOD3 code uses a full non-homogeneous, non-equilibrium, six-equation, two-fluid model for transient simulation of two-phase system behavior. User defined heat structures are used to simulate the reactor fuel rods. Heat transfer coefficients are computed as appropriate for the channel flow and fluid state. A space independent reactor kinetics model is available for reactivity transients.

Some of the RELAP5/MOD3 features important for simulating a natural circulation reactor like MNRC include:

- ability to compute the system density distribution and the gravity force terms in the momentum equation
- ability to implicitly compute the local pool or convective sub-cooled boiling which is known to occur in TRIGA reactors
- a new critical heat flux correlation based upon an extensive tabular set of experimental data
- temperature dependent material properties
- special cross flow models which allow simulation of the two dimensional flow due to radial power differences in the core

While no references to application of the RELAP5 code for analysis of a TRIGA reactor could be found, analyses of many different systems have been reported in the open literature. Many of the system transients analyzed were at low pressure and with natural circulation flow. The RELAP5 code selects the heat transfer correlation to be used based upon the wall temperature and local flow and fluid state. The critical heat flux correlation also uses local conditions and implicitly accounts for axial power distribution. The critical heat flux correlation is further corrected for potential errors if the correlation is entered with flow and fluid conditions which are not in the dominant regions of the data base. The critical heat flux obtained from the tables is also corrected for application to a rod bundle geometry as appropriate. The RELAP5 code can thus be used for analysis of the MNRC thermal and hydraulic performance.

2.0 Reactor Description

The MNRC reactor is a standard design, natural-convection-cooled TRIGA reactor with the graphite radial reflector modified to accept the source ends of four neutron radiography beam tubes. These beams terminate in four separate neutron radiation bays. The reactor core is located near the bottom of a water filled aluminum tank 2.29 m (7.0 ft) in diameter and about 7.47 m (24.5 ft) deep. The water provides adequate shielding over the surface of the tank. The reactor can be operated in a steady state mode by either manual or automatic control. The reactor can also be operated in a square wave or pulsed mode.

The MNRC provides McClellan Air Force Base with the capability to radiograph a wide variety of aircraft components. The facility includes four radiography bays and consequently four beams of neutrons for radiography purposes. All bays contain the equipment required to position the aircraft parts for inspection. The system is designed to operate 24 hours per day.

To continue to achieve a high utilization factor for the facility, additional missions are being identified. These include, but are not necessarily limited to, the following:

- Examination of advanced design turbine blades, both military and commercial.
- Boron neutron capture therapy (BNCT) research for which an epithermal flux is needed.
- Neutron irradiation of silicon-based solid state materials to improve their properties. This application requires a very thermalized flux.

The TRIGA fuel is characterized by inherent safety, high fission product retention, and the demonstrated ability to withstand water quenching with no adverse reaction from temperatures to 1100°C (2012°F). The inherent safety of this TRIGA reactor has been demonstrated by the extensive experience acquired from similar TRIGA systems throughout the world.

The nominal operating parameters for the MNRC TRIGA reactor operating at 2.0MW are presented in Table 1.

3.0 Analysis

As power in the MNRC core is increased, nucleation begins to occur on the fuel rod surfaces and fully developed nucleate boiling occurs. If the surface heat flux remains below the critical heat flux (CHF) it is possible to increase the heat flux without an appreciable increase in fuel rod surface temperature. If the CHF is exceeded, film boiling occurs and the surface temperature increases almost immediately to a much higher values and fuel rod damage will occur. The safe operation of the reactor is dependent upon the operating heat flux in relation to the critical heat flux. The ratio of the critical heat flux to the peak core heat flux is a measure of the safety margin. A critical heat flux ratio (CHFR) of 2.0 is normally assumed for a safety limit. In addition, a steady state peak fuel temperature limit of 750°C (1382°F) is used.

A phenonomon refered to as "chugging" has been observed to occur in operating TRIGA reactors (Reference 7). If steam bubbles coalesce in the hottest cooling channel to form a void, the negative void coefficient abruptly reduces the reactor power. At the lowered power level, the steam void collapses returning the reactor power to the original level and the process is repeated. The vapor void and liquid subcooling were examined to estimate the potential for "chugging".

Maximum steady-state power	2 MW
Neutron flux	~ 10^{14} n/cm^2-sec
Fuel type	TRIGA
Fuel-moderator material	U-$ZrH_{1.6-1.7}$
Uranium content	8.5 to 20 wt. %
Uranium enrichment	Up to 20% U-235
Length of fuel	38 cm (15 in) overall
Diameter of fuel element	3.75 cm (1.478 in) OD
Diameter of fuel meat	3.65 cm (1.435 in) OD
Cladding material	0.051 cm (0.020 in) 304 SS
Number of fuel elements	~80-120
Excess reactivity	Up to $9.50
Cold-hot reactivity loss	~$4
Reactivity loss for equil. xenon	~$2.7
Number of control rods	4-6
Total reactivity worth of rods	$10-$18
Reactor cooling	Natural convection of pool water

Table 1 Nominal Design Parameters

The RELAP5 model used in the MNRC analyses is shown in Figure 1. The model specifies pipe, branch, or single volume components for all major regions of water between the lower grid plate and the upper water surface. These components are connected by junctions as required. Heat structures are defined to simulate the fuel in the average core and hot channel. The hot fluid channel is conservatively assumed to be connected only to rods with the hottest fuel rod power. Pipe components are divided into a user specified number of volumes. In the core region where the axial distribution is important, pipe components with 9 axially distributed volumes were used for the average and hot channel regions. Branch components contain a single volume with a user specified number of junctions connecting to other components. Branches were used to model the unfueled rod regions directly above and below the active core. Single volume and single junction components were used to model the balance of the system.

The primary loop including the N-16 diffuser was also modeled. Time dependent and single junctions were used to model the flow from the upper reactor tank and the return flow to the diffuser region and lower tank. The diffuser flow was assumed to be 20% of the total primary flow. A time dependent volume was used to reference the entire model to atmospheric pressure.

The net driving force for flow within the MNRC tank is the difference between the net buoyancy of the water heated in the core and the friction within the flow paths. Both are computed implicitly by the RELAP5 code. The friction losses consist mainly of the wall friction within the fuel pin flow channels and form losses in the upper and lower grid regions. Friction in other flow paths are computed but are small due to the low velocities. The wall friction is computed directly within RELAP5. The form loss coefficients for the upper and lower grid regions are supplied as input to the code and were computed from data presented in handbooks for similar geometries. The calculated loss coefficients are significantly larger than those used by General Atomics (Reference 5) in their analyses. The conservative computed values were used for the reactor thermal and hydraulic analyses.

RELAP5Components

102 tank outside core barrel(11 volumes)
105 non fueled rod region
106 average core region(9 volumes)
107 hot channel region(9 volumes)
108 upper non fueled rods
109 upper tank over diffuser (3 volumes)
110 connecting junction
120 N16 diffuser region
121 connecting junction
122 time dependent diffuser temperature
123 diffuser injection junction
124 time dependent pressure volume
125 primary outlet flow
128 connecting junction
129 primary return temperature

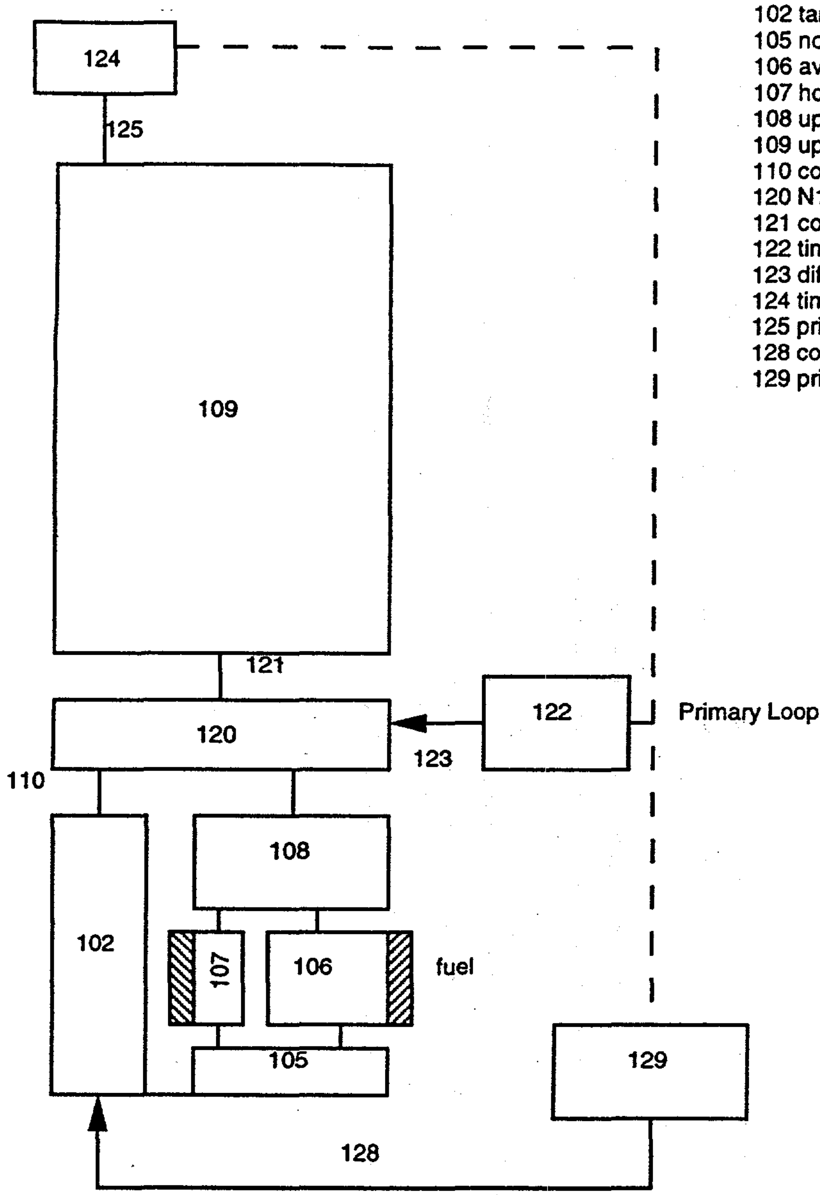

Figure 1 MNRC RELAP5 Model

The steady state fuel temperature depends strongly upon the thermal resistance at the fuel cladding interface. The resistance was assumed to be zero as in prior analyses (Reference 1).

The buoyancy of the water in the core hot channel can be influenced by the cross flow between the hot and average channels. Traditionally the hot and average channels have been assumed to be completely separate (no cross flow) because of the very narrow spacing between the fuel rods. The RELAP5 code provides a means for estimating the effects of cross flow between the hot and average flow channels. The cross flow effect is expected to be very small, and it is impossible to assess the accuracy of computed cross flows. Scoping calculations with RELAP5 showed cross flow to have no effect on fuel temperature and to slightly increase the critical heat flux ratio. Thus, cross flow is conservatively neglected in this analysis.

Benchmarking consisted of comparing the RELAP5 results to measured fuel temperatures and to the results of calculations performed by others. Core loading data and measured fuel temperatures for a Bangladesh reactor of a similar design as MNRC were obtained from General Atomics. The RELAP5 calculated fuel temperature of 425 °C compared very well with the measured temperature of 415 °C. The RELAP5 results were also found to compare favorably with values reported in the Safety Analysis Report for a nearly identical reactor, Torrey Pines (Reference 6).

3.1 Steady State Results

The RELAP5 model described above was used to evaluate the thermal and hydraulic performance of the MNRC during steady state operation at a power level of 2.0MW. The power distribution in the model corresponds to worst case conditions. This is a loading with the control rods lowered 1/3 of their travel from the full up position. The axial peaking factor was 1.33. The core was assumed to have 101 fuel elements with the hot fuel rod operating at 33.2kW for a radial peaking factor of 1.68. The total peaking factor (axial * radial) was 2.23 which is higher than the 2.0 assumed in prior SAR analyses. The radial power distribution in the fuel was assumed to be uniform. The temperature dependent fuel thermal properties were obtained from Reference 4. Two calculations were performed. The power level and core inlet temperature were assumed to be at the limiting operating power of 2.3 MW and 35°C respectively for the first case and at nominal operating conditions of 2.0 MW power and 32.2°C inlet temperature for the second case.

The steady state results are presented in Table 2.

The minimum critical heat flux ratio of 2.51 is much higher than the values calculated in prior safety analyses because of the very conservative correlations used in the past. The current value indicates that a significant margin exists between the proposed operating power(2MW_t) and the power which would result in exceeding the critical heat flux. The magnitude of the critical heat flux is dependent upon local fluid conditions as well as channel inlet conditions and power. The change in magnitude as power increases is, thus, not linear with power and the critical heat flux correlation cannot be used directly to determine the CHFR. It is not practical to perform numerous RELAP5 analyses to determine the power level at which the CHFR would exactly equal 1.0 as was done in prior analyses(Reference 1). An alternate approach was chosen in which a calculation was performed to show that film boiling will not occur at a power significantly above the new operating power. A calculation at 3.0 MW_t resulted in a CHFR of 2.0 and a maximum fuel temperature of 870°C. This calculation demonstrates that even when operating at power which results in the steady state fuel temperature above the 750°C limit, some CHFR margin exists. Operation at 3.0MW_t is clearly not acceptable even though film boiling is not predicted to occur. The predicted outlet fluid temperature at 3.0 MW_t was at saturation and a void fraction of approximately 15% occurred. This is expected to result in fluid channel "chugging" which is known to cause power fluctuations in TRIGA reactors.

All other reactor parameters in Table 2 are acceptable. The predicted fuel temperature is well below the 750 °C limit.

The calculated axial fluid temperature and void distribution are shown in Figure 2. The calculated coolant outlet temperature is subcooled and, thus, the predicted voids are expected to condense immediately after detaching from the fuel rod surface. Chugging and the resultant power fluctuations are, therefore, not expected to occur. If the power fluctuations were to occur, they are not a safety consideration and could be eliminated by adjusting the reactor operation.

Parameter	At Limiting Inlet Temperature (35°C) and Power (2.3MW)	At Nominal Inlet Temperature(32.2 °C) and Power (2.0MW)
Diameter of Fuel Element	3.75 cm	3.75 cm
Length of Fuel Element	38.1 cm	38.1 cm
Flow Area	546 cm2	546 cm2
Hydraulic Diameter	1.86 cm	1.86 cm
Heat Transfer Surface Area	4.53 m2	4.53 m2
Inlet Coolant Temperature	35.0 °C	32.2 °C
Exit Coolant Temperature	106°C	103 °C
Upper Pool Temperature	66°C	57 °C
Coolant Mass Flow	7.7 kg/sec	6.7 kg/sec
Avg Fuel Temperature	373°C (hot pin) 273°C (average pin)	341°C (hot pin) 254°C (average pin)
Maximum Clad Surface Temperature	146 °C	144 °C
Maximum Fuel Temperature	705 °C	630 °C
Avg Heat Flux	50.8 w/cm2	44.2 w/cm2
Max Heat Flux	113w/cm2	98 w/cm2
Hot Channel Outlet Void	4.0%	2.0%
Core outlet subcooling	8 °C	11 °C
Minimum CHF Ratio	2.51	2.94

Table 2 Steady State Results

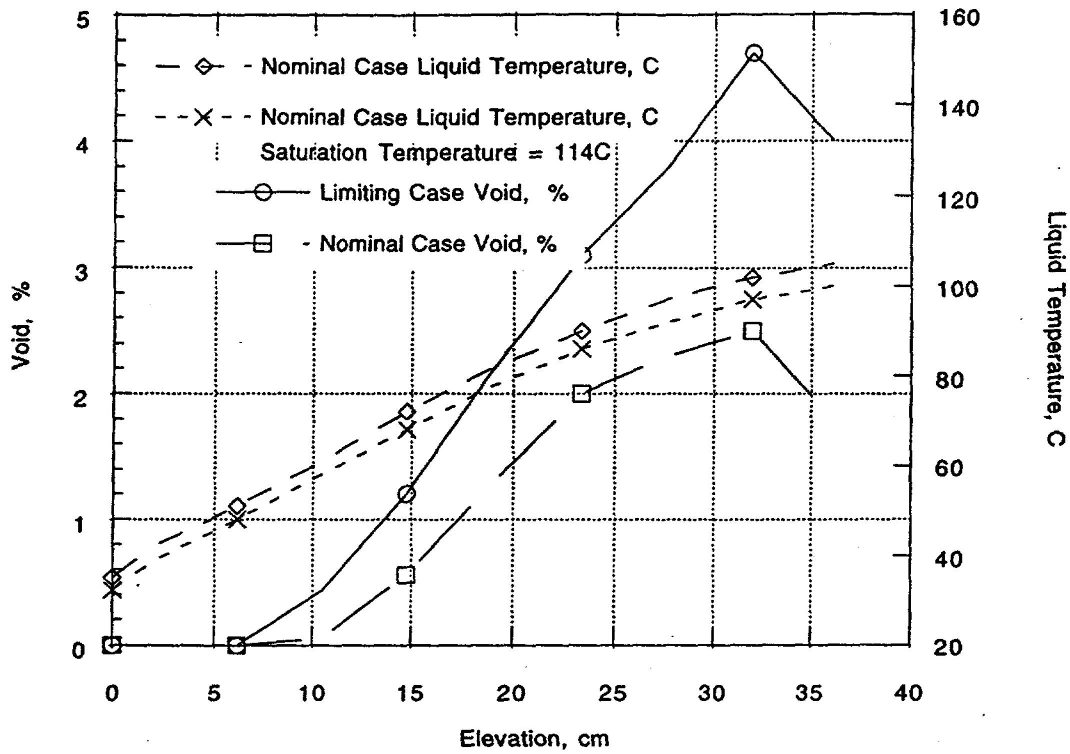

Figure 2 Steady State Temperature and Void Distribution

4.0 Conclusions

The RELAP5/Mod3.1 computer program has been used to successfully perform thermal-hydraulic analyses to support the Safety Analysis for increasing the MNRC reactor from 1.0 MW to 2.0 MW. The calculation results show the reactor to have operating margin for both the fuel temperature and critical heat flux limits. The calculated maximum fuel temperature of 705 °C is well below the 750 °C operating limit. The critical heat flux ratio was calculated to be 2.51.

5.0 References

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