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EVALUATION (USING A HYBRID COMPUTER) OF A SHEATH
TYPE ELECTRICAL HEATER ROD FOR SIMULATING
THE RESPONSE OF A NUCLEAR FUEL ROD

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ABSTRACT

A sheath type electrical heater rod was evaluated to determine how well the surface temperature and heat flux responses of the heater rod would simulate those of a nuclear reactor fuel rod exposed to the conditions expected to occur during a postulated loss-of-coolant accident for a pressurized water reactor.

The study was conducted by modeling a single axial level of the sheath type heater and nuclear rod with a hybrid computer and imposing boundary conditions at the cladding surface. A coolant model was also included to predict boundary conditions based on the surface heat flux and local coolant conditions. A filament type rod previously used in experiments was also modeled with boundary conditions imposed at the cladding surface.

The evaluation showed that the power input to a sheath type heater rod can be predetermined so that when the sheath type heater rod is exposed to the same conditions as a nuclear rod the surface temperature and heat flux of the sheath type heater rod will be almost the same as those of the nuclear rod. The evaluation also showed that the sheath heater rod has greater potential for simulating the nuclear rod than does the filament type heater rod. Since other factors, such as power, mechanical, and operational limitations, measurement techniques, and experimental objectives, not included in this investigation must be considered prior to the selection of a heater rod design for a particular experiment, no recommendations as to specific application are made.

SUMMARY

Electrically heated rods are commonly used to simulate nuclear fuel rods for experiments designed to obtain thermal-hydraulic data applicable to large water cooled reactors during steady state and transient conditions. The heater rods when exposed to the conditions to which the nuclear fuel rod would be exposed should provide the same surface temperature and heat flux as would be obtained for the nuclear fuel rod. This report presents the results of an evaluation of a particular design and operating technique of a sheath type heater rod. To evaluate this design and operating technique, analytical models were developed for the heat transfer processes for a sheath type heater rod, a nuclear rod, and a filament type heater rod. A coolant channel was included in the modeling of the sheath type and nuclear rods to calculate boundary conditions based on the surface heat flux and the local fluid conditions. Comparisons were made among the responses of the heater rods and the nuclear rod when the heater rods were exposed to conditions expected to occur for the nuclear rod during a postulated loss-of-coolant accident in a large pressurized water reactor. The study was conducted by modeling a single axial level of the rods with a hybrid computer and imposing boundary conditions at the cladding surface. The nuclear rod was simulated better by the sheath type heater rod than by the filament type heater rod. However, since other factors, such as power, mechanical, and operational limitations, measurement techniques, and experimental objectives not included in this investigation must be considered prior to selection of a heater rod design for a particular experiment, no recommendations as to specific application are made.

A hybrid computer was used for this heater rod evaluation. The results from this analysis technique were nearly identical with results from digital computer codes. In addition, the direct involvement of the analyst in the hybrid computation process permitted rapid evaluation and comparison of the models as they were developed and implemented.

NOMENCLATURE

A	Cross-sectional area (ft^2)
C_p	Specific heat ($\text{Btu}/\text{lb}_m \cdot ^\circ\text{F}$)
D_e	Equivalent diameter (in.)
G	Mass flux ($\text{lb}_m/\text{ft}^2\text{-sec}$)
h'	Heat transfer coefficient ($\text{Btu}/\text{ft}^2\text{-sec-}^\circ\text{F}$)
h	Thermodynamic enthalpy (Btu/lb_m)
k	Thermal conductivity ($\text{Btu}/\text{ft-sec-}^\circ\text{F}$)
K	Thermal conductance ($\text{Btu}/\text{sec-}^\circ\text{F}$)
ℓ	Length between nodal locations (ft)
M	Mass of nodal volume (lb_m)
$\dot{m}$	Mass flow (lb_m/sec)
P	Pressure (psia)
Pr	Prandtl number
q	Latent heat for phase change (Btu/lb_m)
q''	Heat flux ($\text{Btu}/\text{ft}^2\text{-sec}$)
q'''	Power density ($\text{Btu}/\text{ft}^3\text{-sec}$)
Q	Volumetric flow rate (ft^3/sec)
$\dot{Q}$	Heat transfer per unit time (Btu/sec)
r	Radius (ft)
R	Radius of fuel rod (ft)
S	Integration surface area [Equation(2)]
t	Time (sec)
T	Temperature ($^\circ\text{F}$ or $^\circ\text{R}$)
V	Volume (ft^3)
v	Specific volume (ft^3/lb_m)
X	Thermodynamic quality

GREEK LETTERS

α	Coefficient of linear expansion (1/°F)
ρ	Density (lb _m /ft ³)
μ	Viscosity (lb _m /ft-sec)

SUBSCRIPTS

avg	Average
c	Refers to channel
clad	Refers to cladding
cs	Refers to cladding surface
CHF	Critical heat flux
eff	Refers to effective thermal conductivity
f	Refers to saturated liquid conditions
fg	Refers to phase change from saturated liquid to saturated vapor
fs	Refers to fuel surface
fuel	Refers to fuel
g	Refers to saturated vapor conditions
gap	Refers to gap between fuel and cladding
He	Refers to helium gas in gap
i, j	Radial node number
in	Refers to channel inlet conditions
m	Refers to material
o	Refers to initial ambient condition
out	Refers to channel outlet conditions
s	Refers to interface between regions
sat	Refers to saturation conditions
surf	Refers to surface conditions

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EVALUATION (USING A HYBRID COMPUTER) OF A SHEATH TYPE ELECTRICAL HEATER ROD FOR SIMULATING THE RESPONSE OF A NUCLEAR FUEL ROD

I. INTRODUCTION

Electrically heated rods are commonly used in experiments as the heat source simulating uranium dioxide (UO_2) fuel rods typical of those used in light-water reactor cores. These experiments are designed to provide data for development and evaluation of analytical models that describe the thermodynamic and hydraulic behavior of the reactor coolant and core and mechanical behavior of the structural system during blowdown in a postulated loss-of-coolant accident (LOCA). The experiments are conducted to gain information on the coolant behavior in the primary system and on the rate of heat transfer to the coolant[1].

Since the coolant behavior is affected by the heat source used to simulate the UO_2 fuel rod, the design and operation of the electrical heater rod should be such that the heater rod behaves similarly to the UO_2 fuel rod. The effect of the heater rod on the coolant will be the same as that of the UO_2 fuel rod if the cladding surface temperature and surface heat flux for the heater rod are the same as for the nuclear rod.

The conditions of fluid flow, pressure, enthalpy, and quality produced during a transient experiment cannot be known precisely before an experiment is conducted. Thus, the heater rod should be designed so that representative temperatures and heat fluxes can be obtained independent of the fluid conditions, thereby avoiding the need for a number of experiments followed by repeated changes in the method of operation to optimize the response of the heater rod.

Heater rods commonly used in experiments utilize filament or sheath electrical resistance elements. Both types of rods have been reasonably satisfactory in fulfilling the objectives of the experiments in which they have been used. However, new developments have occurred in heater rod technology, and these developments should be evaluated to determine whether they will lead to better simulation and thus more nearly complete information. One new type of heater rod which appears promising for providing better simulation of the desired behavior is a sheath heater[2] containing uranium dioxide. In this application the UO_2 acts only as a storage reservoir for thermal energy. This heater rod was investigated to determine how well it would simulate a nuclear rod. Since other factors not included in this investigation must be considered prior to selection of a heater rod design for a particular experiment, no recommendations as to specific application are made.

To accomplish the investigation, a hybrid computer (a digital-analog computer) model was developed to simulate a single axial level in a nuclear fuel rod, the new sheath type heater rod, and a filament type heater rod. The hybrid computer was used because of its capability to handle this type of problem in an efficient rapid manner and because of the ease with which parametric studies can be accomplished. The filament type heater rod was included because this type of rod generally has been used in experimental and analytical programs conducted previously by Aerojet Nuclear Company[3,4]. Sheath type heaters of other designs have been used in other experiments but were not considered because of the

limited scope of this work. A model of the coolant channel associated with a nuclear fuel rod was also developed to calculate coolant properties necessary to determine the proper heat transfer regime and magnitude of the heat transfer coefficient during single phase saturated liquid and two phase flow conditions. The coolant channel was used in conjunction with the nuclear and sheath type heater rod models. Preliminary calculations for the filament type heater rod were sufficient for comparison purposes without including the coolant channel with the model. This report contains a description of the hybrid models used.

Appendices are included which describe the hybrid simulation of the various rod types and the coolant channel, the hybrid computer symbolism and the material properties for the various rod types, and verification that the sheath type heater rod design and operating technique will simulate the nuclear rod.

The investigation was performed and is described in the following order. First, a hybrid model was developed for the nuclear rod. The solution obtained for the temperature distribution in the rod was compared with results obtained from digital computer code calculations from previous studies[4]. Second, a model of the sheath type heater was developed and the response calculated by the model was compared to the response calculated by the nuclear rod model. A coolant channel model was then added to the nuclear and sheath type heater models and additional comparisons were made. A model of a filament heater rod was then developed and the response calculated by the model was compared with the response calculated by the model of the nuclear rod without the coolant channel.

II. HYBRID MODEL OF A NUCLEAR ROD

A hybrid model was developed to describe conduction heat transfer in the radial direction with internal power generation for a UO₂ zircaloy-clad, cylindrical nuclear fuel rod. The model included a thermal resistance for the fuel-cladding gap and a cladding surface-to-coolant film heat transfer coefficient. A continuous-time, discrete space method of solution of the heat diffusion equation was used to allow time control of the power generation rate as for the electrically heated rod. The model was developed with five nodal volumes in the fuel and one in the cladding because this nodal pattern was shown to be adequate by previous studies[5]. Although the development of the model in this investigation is based on one axial level, the simulation could be extended to include additional axial levels if the magnitude of the axial conduction component is minor in comparison to the radial conduction component as is true in most applications for LOCA analysis. The axial levels then could be multiplexed or stacked[6]. Under these circumstances a single axial level could be programmed and the outlet conditions recorded as functions of time. These time functions then would become the inlet boundary conditions for the next downstream level until the length of the channel was traversed. This technique would allow a rapid analysis of a transient because the hybrid solution of any given axial level would be carried out in real time or faster.

The geometrical configuration, simulation techniques, and power functions used in this investigation for the nuclear rod are described in Appendix A. The thermal physical properties used are described in Appendix B.

A volume integration of the heat diffusion equation was performed and the energy balance equation was written for each of the nodal volumes as:

$$\int_V \rho C_p \frac{\partial T}{\partial t} dV = \int_V q''' dV - \int_V \vec{\nabla} \cdot k \vec{\nabla} T dV \quad (1)$$

where ρ is the density, C_p is the specific heat at constant pressure, T is the temperature, t is the time, V is the volume, q''' is the power density, $\vec{\nabla}$ is the vector operator, and k is the thermal conductivity. The nomenclature used throughout this report is listed at the front of the report.

A volume-weighted average temperature was defined for each nodal volume. Heat generation was assumed constant in the radial direction and generated at the nodal volume center, or nodal point. The volume integral of the divergence term was converted to a surface integral by applying the divergence theorem:

$$\int_V \vec{\nabla} \cdot k \vec{\nabla} T dV = \int_S \vec{n} \cdot k \vec{\nabla} T dS \quad (2)$$

where $\vec{n}$ is a unit vector normal to the surface S . The resulting equation in one dimension for a nodal volume of the fuel and cladding is:

$$\left(\rho V C_p \frac{dT}{dt} \right)_i = \left(V q''' \right)_i + \sum_{n=1} \left(\frac{kA}{\ell} \right)_i \left(T_n - T_i \right) \quad (3)$$

where n represents one of the two surfaces for the i -th node, and ℓ is the distance from the center of the nodal volume to the surface. Figure 1 shows an example of a nodal volume. Continuity of heat flux is expressed at a surface common to two nodal volumes, i and j , as follows:

$$\left(\frac{k}{\ell}\right)_i \left(T_{i-1} - T_n\right) = \left(\frac{k}{\ell}\right)_j \left(T_n - T_j\right) \quad (4)$$

The surface temperatures are eliminated in equations describing the temperature distribution in adjacent nodes. The resulting heat diffusion equation for a nodal volume, i , is as follows:

$$\rho V C_p \frac{dT_i}{dt} = V q_i''' + K_{i-1,i} \left(T_{i-1} - T_i\right) - K_{i,i+1} \left(T_i - T_{i+1}\right) \quad (5)$$

where, for example, the conductance $K_{i-1,i}$ is:

$$K_{i-1,i} = \frac{\left(\frac{kA}{\ell}\right)_{i-1} \left(\frac{kA}{\ell}\right)_i}{\left(\frac{kA}{\ell}\right)_{i-1} + \left(\frac{kA}{\ell}\right)_i} \quad (6)$$

For nodes spaced at equal intervals in the radial directions, the geometrical factor A/ℓ can be factored out leaving:

$$K_{i-1,i} = \left(\frac{k_{i-1} k_i}{k_{i-1} + k_i}\right) \left(\frac{A}{\ell}\right)_i = k_{eff} \left(\frac{A}{\ell}\right)_i \quad (7)$$

At the interface, s , between the fuel and cladding (fuel-cladding gap) or the interface between the cladding and coolant (convective film), continuity is expressed as follows:

$$\left(\frac{k}{\ell}\right)_m A_s \left(T_m - T_s\right) = h' A_s \Delta T \quad (8)$$

Fig. 1 Example of a nodal volume.

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where h' is the gap or film heat transfer coefficient, A_s is the surface area for heat transfer, and ΔT is the temperature difference ($T_s - T_{s'}$) across the gap or convective film. Figure 2 shows the modeling and location of the terms.

The fuel-cladding gap heat transfer coefficient is based on a variable thermal-mechanical expansion model for the fuel and cladding. This model considers the gap to contain helium gas, the conductivity of which is evaluated at the arithmetic average temperature of the gas. The thickness of the fuel-cladding gap is computed by:

$$\ell_{\text{gap}} = \ell_o + \Delta r_{\text{clad}} - \Delta r_{\text{fuel}} \quad (9)$$

where ℓ_o is the thickness of the gap at ambient conditions. The fuel expansion, Δr_{fuel} , is computed for n nodes by:

$$\Delta r_{\text{fuel}} = \frac{2}{r_o} \sum_{i=1}^n \ell_i \alpha_i \left(T - T_o \right)_i r_{i\text{avg}}. \quad (10)$$

The cladding expansion, Δr_{clad} , is computed by:

$$\Delta r_{\text{clad}} = r_o \alpha \left(T - T_o \right). \quad (11)$$

The gap heat transfer coefficient then becomes:

$$h'_{\text{gap}} = k_{\text{He}} / \ell_{\text{gap}}. \quad (12)$$

Fig. 2. Interface modeling.

III. COMPARISON OF HYBRID AND DIGITAL CODE RESULTS FOR NUCLEAR ROD

The results obtained from the conduction model of a nuclear fuel rod developed for the hybrid model are compared with results obtained from digital models using the codes, HEAT-1[7] and THETA1-B/MOD 001[5].

Figure 3 shows a comparison of the results obtained from the six-radial node hybrid model[a], with those obtained from the detailed 30-radial node model used in HEAT-1[4]. The agreement between the results for nodes located at the same radial position is excellent with the difference between values predicted by the two models being less than one percent.

Fig. 3 Steady state radial temperature profile for HEAT-1 and hybrid models -- nuclear fuel rod.

[a] Figure 3 and subsequent figures indicate nine locations at which temperatures were calculated: one at each of the five nodes in the fuel, one at the single node in the cladding, and one each at the fuel surface, inner cladding surface, and outer cladding surface. The subscripts one through six on T indicate the nodes.

The thermal-physical properties used were the same for both models. In both models the heat capacity of a nodal volume was based on the nodal temperature, and the thermal conductivity between nodes was based on a linear average of the thermal conductivities computed for each of the nodes. The temperatures at the cladding surface and at the innermost node obtained from the hybrid model are compared in Figure 4 with those obtained from the HEAT-1 code for a transient LOCA. The cladding-to-coolant heat transfer coefficient applied is shown in Figure 5. The critical heat flux is exceeded at initiation of the LOCA. The coolant sink temperature is shown in Figure 6. The agreement shown in Figure 4 for the results from the hybrid model with results from the HEAT-1 model is within two percent or better, indicating that a six-node hybrid model is sufficiently detailed to simulate the transient thermal response of the nuclear rod. Previous studies[5] with the THETA1-B code also indicated that the node selection for the hybrid model was sufficient.

The comparison of the results of the hybrid model with those from the THETA1-B code was not as straightforward as with the HEAT-1 results because of basic model differences. The THETA1-B code has provision for several axial levels. Also, THETA1-B computes a surface-to-coolant film heat transfer coefficient based on coolant conditions, the rod surface temperature, and the surface heat flux. In the comparison shown in Figure 7, the film heat transfer coefficient and coolant temperature, shown in Figures 8 and 9 as computed by THETA1-B, were used as input quantities to the hybrid model. The axial level simulated on the hybrid computer was equivalent to the hottest axial level in the particular THETA1-B case. The heat fluxes compare favorably except in the regions of rapid changes in the film heat transfer coefficient following the occurrence of the critical heat flux when transition boiling and film boiling occur.

Fig. 4 Comparison of temperatures obtained from HEAT-1 and hybrid models during transient -- nuclear fuel rod.

Fig. 5 Assumed heat transfer coefficient during LOCA -- nuclear fuel rod.

Fig. 6 Assumed coolant sink temperature during LOCA -- nuclear fuel rod.

Fig. 7 Comparison of transient heat flux obtained from THETA1-B and from hybrid model using results obtained from THETA1-B during LOCA -- nuclear fuel rod.

Fig. 8 Surface heat transfer coefficient as computed by THETA1-B and used as input to hybrid model.

Fig. 9 Coolant channel temperature as computed by THETA1-B and used as input to hybrid model.

Part of the difference in the results is due to the difficulty of implementing the film heat transfer coefficient exactly as it was predicted and used in the THETA1-B model. Since the requirement of this comparison was to determine the consistency of the hybrid approach with numerical methods such as used in THETA1-B, the small differences shown in Figure 7 were not eliminated.

This comparison showed that the hybrid method gave results consistent with those obtained using the numerical techniques of digital codes. The hybrid computer was found to be easily programmed and to provide rapid solution for the problems of interest in this study. The direct involvement of the analyst was also found to be an asset in the parametric studies.

IV. DEVELOPMENT OF A HYBRID MODEL OF A SHEATH TYPE HEATER ROD AND COMPARISON OF RESULTS FOR NUCLEAR AND SHEATH RODS

Only minor changes to the hybrid model of the nuclear fuel rod were required to model the sheath type heater rod. The response of the sheath type heater rod was then compared with that of the nuclear rod.

1. MODEL DESCRIPTION

The conduction model developed to describe power generation and heat flow in the sheath type heater rod was essentially the same as that used for the nuclear rod except that the power was generated in the sheath or cladding of the rod and the cladding material was stainless steel. The same equations developed for the nuclear rod were used to describe the sheath type heater rod. The geometrical configuration and simulation techniques are described in Appendix A and the physical properties are described in Appendix B.

The method used to determine the required rate of power generation in the cladding of the sheath type heater rod for simulation of the nuclear rod is described in detail in Appendix C. In this method, proposed originally by the Battelle, Pacific Northwest Laboratories[2], the required rate of power generation in the cladding was calculated and applied as outlined in the following procedure:

- (1) The heat flux from the cladding to the coolant was determined as a function of time for the nuclear rod by maintaining the surface temperature constant and equal to that during steady state operation while reducing the power generation rate according to the decay heat generation expected during the LOCA transient.
- (2) The heat flux from the cladding to the coolant of the nuclear rod was input as the heat source to the sheath type heater rod which was initially at a uniform temperature identical to the cladding surface temperature of the nuclear rod. The electrical heater was then exposed to the LOCA environment by applying the expected effective heat transfer coefficient and sink temperature. The actual surface temperature and heat flux were then calculated for the electrical heater rod.

The surface heat flux and temperatures were also determined for the nuclear rod as functions of time for the same LOCA heat transfer coefficient and sink temperature as used for the sheath type rod. The decay heat generation applied to the nuclear rod was that expected during the LOCA transient. The surface heat fluxes and temperatures obtained compared favorably with those for the heater rod. Further details of the verification of this method are presented in Appendix C.

Since the method of determining the power input for the electrically heated rod was developed for the special case of constant thermal properties and cladding surface heating, further studies were conducted to determine the effects of volumetric cladding heating, variable thermal properties, and a variable gap heat transfer coefficient. Transients of interest in this study were those resulting from an LOCA, thus film heat transfer coefficients and channel coolant temperatures typical of those for nuclear fuel rods computed by the digital code THETA1-B were used as boundary conditions for the hybrid model.

2. COMPARISON OF RESULTS FOR NUCLEAR ROD AND SHEATH TYPE HEATER ROD

The following results were obtained by the procedures outlined for the case when the power is generated volumetrically in the cladding and the effects of variable thermal properties and a variable gap coefficient are considered. The analytically determined electric power function in Figure 10 is the power generation rate used for the sheath type heater rod. The comparisons of surface heat flux shown in Figure 11 and the cladding surface temperatures shown in Figure 12 were obtained. The film heat transfer coefficient (Figure 7) and coolant temperature (Figure 9) were used as boundary conditions for the LOCA transient. As can be seen in Figures 11 and 12, the difference between the response of the sheath type heater rod and the nuclear rod becomes sizeable after CHF occurs.

Also shown in Figures 11 and 12 are the results for a sheath type heater rod for which the gap heat transfer coefficient was maintained constant at $10,000 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F}$, which is a representative value for a rod of swaged construction. This constant coefficient resulted in a lower temperature after CHF and indicated the analytically determined electrical power function was too small to maintain the surface temperature of the sheath type rod equal to that of the nuclear rod for the same LOCA transient.

Since the surface temperatures of the nuclear rod and the sheath type heater with a variable gap heat transfer coefficient were not the same for the heat generation rate determined by the method previously described, the electrical power function needed to make them the same for the particular boundary conditions shown was computed by the implicit solution technique described in Appendix A. For this case the cladding surface temperature of the sheath type rod with a variable gap heat transfer coefficient was compared to that of the nuclear rod, and the difference between the two was used to derive the power input to the sheath rod.

Fig. 10 Electrical heat generation rate for hybrid model.

Fig. 11 Surface heat flux for nuclear rod and sheath type rod during transient.

Fig. 12 Cladding surface temperature for nuclear rod and sheath type rod during transient.

The resulting electrical power function is compared in Figure 13 with the analytically determined power function previously shown in Figure 10. The differences between the curves in Figure 13 are attributed to the temperature effect of the thermal properties and finite cladding thickness of the sheath type heater rod.

Fig. 13 Comparison of electrical heat generation rates for hybrid model.

V. DEVELOPMENT OF MODEL FOR HEAT TRANSFER TO THE COOLANT AND COMPARISON OF RESULTS FOR NUCLEAR AND SHEATH RODS

In order to determine whether the differences in heat flux and cladding surface temperature, shown in Figures 11 and 12, for the nuclear rod and sheath type rod might be larger under actual operating conditions, a more realistic model for heat transfer to the coolant was added to the simulation. A better simulation of the heat transfer was desired because the surface heat transfer coefficient was not independent of the thermodynamic state and mass flux of the coolant in the channel as was implied in the model development up to this point.

The following sections describe the coolant model and the heat transfer correlations and presents a comparison of the results for the sheath type heater rod with those for the nuclear rod using this coolant model.

1. COOLANT MODEL

The basic model developed to simulate the heat transfer to the coolant associated with a heated rod consisted of the one-dimensional conservation of energy equation for two phase flow, and the conservation of mass equation for a single control volume. A continuous time, discrete space, method of solution of these equations was used[a]. The model was limited to single phase saturated liquid and two phase flow because of the lack of space in the digital equipment for storage of superheated steam property tables.

The following basic assumptions were made to solve these conservation equations:

- (1) Thermodynamic equilibrium exists between liquid and vapor phases
- (2) The two phases travel at the same velocity
- (3) The pressure is spatially constant throughout the control volume and equal to the channel inlet pressure
- (4) Turbulent and viscous effects are negligible.

The conservation equations were then written in the following form for the channel control volume:

$$\frac{dMh_c}{dt} = \dot{m}_{in} h_{in} - \dot{m}_{out} h_{out} + \dot{Q}_{surf} \quad (13)$$

[a] An alternative method of simulation of coolant channels is a serial approach called the continuous space, discrete time method[8]. In this method the coolant channel length is the continuous variable and time is separated into discrete increments. Such a method has proved useful in the simulation of tube steam generators[9].

and

$$\frac{dM}{dt} = \dot{m}_{in} - \dot{m}_{out} \quad (14)$$

where $\dot{Q}_{surf}$ represents the heat being transferred from the rod surface to the coolant. A linear enthalpy rise

$$h_c = \left(h_{in} + h_{out} \right) / 2 \quad (15)$$

was assumed across the control volume.

Substitution of the expression for h_{out} from Equation (15) into Equation (13) results in the final form of the energy equation:

$$\frac{dMh_c}{dt} = \dot{m}_{in} h_{in} - \dot{m}_{out} \left(2h_c - h_{in} \right) + \dot{Q}_{surf} \quad (16)$$

The inlet coolant enthalpy, flow, and pressure to the control volume adjacent to the axial hot spot obtained from RELAP3[10] output which gives the values at the plenum-core junction were applied to THETA1-B to obtain the proper values at the specific axial location.

The implementation of these equations to provide values for the heat transfer model and the state equations for water are described in Appendix A.

2. HEAT TRANSFER MODEL

The heat transfer model for the interface between the rod surface and the coolant was developed by assuming that the following regimes of heat transfer could occur during an LOCA: (1) forced convection to a subcooled liquid only, (2) nucleate boiling, (3) transition boiling, and (4) stable film boiling. A critical heat flux (CHF) correlation was programmed to determine the end of nucleate boiling and the onset of transition boiling.

The correlations selected to describe the heat transfer regimes were the following:

Subcooled Forced Convection

Dittus-Boelter[11]

$$q'' = h' (T_{surf} - T_{coolant}) \quad (17)$$

$$h' = (0.023) \left(\frac{12 k}{D_e} \right) Pr^{0.4} \left(\frac{G D_e}{12 \mu} \right)^{0.8} \quad (18)$$

Nucleate Boiling

Thom[12]

$$q'' = \left[\frac{\exp(P/1260)}{4.32} \left(T_{surf} - T_{sat} \right) \right]^2 \quad (19)$$

Transition Boiling

McDonough, Milich, and King[13]

$$q'' = q''_{CHF} - C \left(T_{surf} - T_{surf, CHF} \right) \quad (20)$$

where C is a constant related to the pressure as:

P	C
2000	0.306
1200	0.328
800	0.417

Stable Film Boiling

Dougall and Rohsenow[14]

$$q'' = h' (T_{surf} - T_{sat}), \quad (21)$$

$$h' = 0.023 \left(\frac{12 k_g}{D_e} \right) \left(\frac{Pr}{g} \right)^{0.4} \left[\left(\frac{\rho_g D_e}{12 \mu_g} \right) \left(\frac{Q_f + Q_g}{A_{flow}} \right) \right]^{0.8} \quad (22)$$

Critical Heat Flux

Babcock & Wilcox Company, B&W-2[15]

$$q''_{CHF} = \left\{ \frac{1.15509 - 0.40703 D_e}{(3600)(12.710) \left[3.0545 \left(\frac{3600G}{10^6} \right)^a \right]} \right\} \left\{ (0.3702 \times 10^8) \left[0.59137 \left(\frac{3600G}{10^6} \right)^b - 0.15208 X_{CHF} h_{fg} (3600G) \right] \right\} \quad (23)$$

where a and b are constants related to the pressure as:

$$a = 0.71186 + (0.20729 \times 10^{-3}) (P - 2000)$$

$$b = 0.83040 + (0.68479 \times 10^{-3}) (P - 2000) .$$

The selection of the heat transfer regime was based on a typical flow boiling curve as shown in Figure 14. The solid line represents the map of physically possible heat fluxes and the dashed lines represent the extensions of the analytical correlations.

Fig. 14 Flow boiling curve.

The following procedures were used to select the heat transfer value:

- (1) The heat fluxes for subcooled forced convection and nucleate boiling correlations were calculated and the larger of the two values was selected.
- (2) If the nucleate boiling heat flux was selected and was less than the calculated critical heat flux, the nucleate boiling value of heat flux was used.
- (3) If the nucleate boiling heat flux was larger than the critical heat flux, the heat fluxes for transition boiling and stable film boiling correlations were calculated and the larger of the two values was selected.

The computer logic for selection of the heat transfer regime is described in Appendix

A.

3. COMPARISON OF RESULTS FOR NUCLEAR ROD AND SHEATH TYPE HEATER ROD

The results shown in Figure 15, for the coolant model added to the basic conduction models for both the nuclear rod and sheath type heater rod indicates that the nuclear rod can be simulated closely with the sheath type heater rod. This capability is shown by the agreement between the surface temperatures, the surface heat fluxes, and the critical heat fluxes computed for the two models. The transient conditions of mass flux and pressure, shown in Figure 16, and inlet enthalpy, shown in Figure 17, were chosen to force the calculation to remain in the two phase region for which the correlations used would be applicable. The electric power function used was determined by the method described in Section IV, and is shown in Figure 10. An important result was that small differences in the heat fluxes and surface temperatures early in the transient do not significantly affect the comparison later in the transient. An interesting fact is that CHF was exceeded early in the transient causing film boiling to occur. Later the heat flux needed to maintain film boiling increased, exceeding the actual heat flux so that nucleate boiling was reestablished.

Fig. 15 Response of nuclear rod and sheath type rod during transient -- coolant model included in hybrid simulation.

Fig. 16 Mass flux and pressure input to coolant model during transient.

Fig. 17 Coolant enthalpy input to coolant model during transient.

VI. DEVELOPMENT OF HYBRID MODEL OF A FILAMENT TYPE HEATER ROD AND COMPARISON OF RESULTS FOR NUCLEAR AND FILAMENT RODS

Previous work[4] has indicated that a filament type heater rod is difficult to operate readily in such a manner as to duplicate the surface temperature and heat flux of a nuclear rod when the heater rod was exposed to the same boundary conditions as the nuclear rod. Since filament type heater rods have been developed and are being used for the simulation of nuclear rods[3], additional investigations of power control were considered warranted. Thus, a model of a filament type rod was developed for the hybrid computer to study the response of the filament type rod for various power control boundary conditions. An attempt was made to determine whether an electrical power function existed such that when applied to the filament type rod the surface temperature and heat flux of the filament type rod would duplicate the surface temperature and heat flux of a nuclear rod when the filament rod was exposed to the same boundary conditions as the nuclear rod.

1. MODEL DESCRIPTION

The conduction model developed to describe power generation and heat flow in the filament type heater rod was similar to that developed for the nuclear rod with power generated internally in a spirally wound, filament type heater. The development of the heat diffusion equations was the same as that for the nuclear rod. The geometric configuration and simulation techniques are described in Appendix A.

2. COMPARISON OF RESULTS FOR NUCLEAR ROD AND FILAMENT TYPE HEATER ROD

Previous studies showed that an exact function of electric power versus time required to simulate a nuclear rod with a filament type heater rod could not be determined analytically because the function is dependent on the system hydraulic response and can only be found after the hydraulic response is known and can be fed back into the model. To approximate the electric power input, three methods of controlling the electrical power input to the filament type heater rod were investigated for the LOCA transient boundary conditions shown in Figures 8 and 9.

The first electrical power function used as input to the filament type heater rod was the electrical power function (Figure 10) determined analytically for the sheath type heater rod. The surface temperature and heat flux for the filament type rod obtained through use of this power function are compared in Figures 18 and 19 with the corresponding surface temperature and heat flux for the nuclear rod. During the period when the surface heat transfer coefficient is very large, a small difference in the cladding surface temperature results in a very large difference in the surface heat flux.

The second power function used for the filament type rod was the same power function as used for the nuclear rod. The method of power control is discussed in Appendix A. The resulting surface temperatures and heat flux comparisons are shown in Figures 20 and 21.

The implicit solution technique was also used in an attempt to determine a suitable power function. The variable used for comparison was the surface heat flux. In this

Fig. 18. Comparison of cladding surface temperature of nuclear rod and filament type rod using programmed power as determined for sheath type rod.

Fig. 19 Comparison of surface heat flux of nuclear rod and filament type rod using programmed power as determined for sheath type rod.

Fig. 20 Comparison of cladding surface temperature of nuclear rod and filament type rod using programmed power as determined for nuclear fuel rod.

Fig. 21 Comparison of surface heat flux of nuclear rod and filament type rod using programmed power as analytically determined for nuclear fuel rod.

comparison, the surface heat flux for the nuclear rod was compared to the surface heat flux for the filament type rod for the case in which both rods were exposed to the same boundary conditions, and the difference between the two was used to adjust the power input to the filament type rod. The resulting power function is shown in Figure 22. The surface temperature and heat flux of the filament heater rod using the power function are compared in Figures 23 and 24 with those for the nuclear rod. The exact simulation of the nuclear rod by the filament heater rod would be difficult to attain. The basic difference in the behavior of the two rods is attributed to the heat capacity of the filament type heater rod which causes a time delay between changes in the electrical power input and the surface heat flux response to the electrical power input. However, the results of previous studies indicate that maintaining the filament type rod at full power for a period of time after initiation of the transient would likely minimize differences between the filament and nuclear rods.

Fig. 22 Electrical heat generation rate for filament type rod.

Fig. 23 Comparison of cladding surface temperature of nuclear rod and filament type rod using programmed power as determined by the implicit solution technique.

Fig. 24 Comparison of surface heat flux of nuclear rod and filament type rod using programmed power as determined by the implicit solution technique.

VII. CONCLUSIONS

The results of this study have shown that the electrical power function necessary for a sheath type rod to simulate the response of a nuclear rod can be approximated favorably independent of boundary conditions. Exact simulation by the sheath type heater concept is not possible because of differences between the characteristics of real materials and the mathematical assumptions of the concept. The results also indicate that a sheath heater rod has greater potential for simulating a nuclear rod under LOCA transient conditions than does the filament heater rod considered. However, previous studies have indicated that a filament rod has the capability of providing a proper heat source such that the appropriate hydraulic behavior during blowdown of a large PWR can be obtained in a small system experiment. The potential exists also for the filament rod to provide information for evaluation of the time-to-CHF, post-CHF heat transfer, and reflooding behavior with the aid of interpretive analysis. Additional analysis is required to determine the exact discrepancy in behavior between the sheath or filament heater and the nuclear rod. This additional analysis could be accomplished by exposing both heater rods and the nuclear rod to the particular coolant conditions expected and computing the surface temperatures and heat fluxes. The importance of differences in heater behavior and nuclear rod behavior to the objectives of any experimental program, and the use of the resultant data for analysis development and assessment could then be evaluated.

The hybrid simulation in continuous time, discrete space, of conductive and convective heat transfer occurring in the system described has been shown to be consistent with the finite differencing techniques of digital codes. This study has shown the hybrid computer to be capable of performing a rapid parametric study of a heater rod and its associated coolant channel.

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APPENDIX A

HYBRID SIMULATION OF NUCLEAR ROD WITH SHEATH AND FILAMENT TYPE HEATER RODS

APPENDIX A

HYBRID SIMULATION OF NUCLEAR ROD WITH SHEATH AND FILAMENT TYPE HEATER RODS

The heat diffusion equation [Equation (5) of the body of this report] was simulated on the hybrid computer with finite differencing in the radial dimension in continuous time. The techniques employed, discussed in succeeding paragraphs, were similar for the nuclear and heater rods. Computer symbolism and the thermal-physical properties of the materials have been included for reference in Appendix B.

1. NUCLEAR ROD

The nuclear rod simulated was typical of PWR fuel rods in physical dimensions and materials. The fuel material was uranium dioxide and the cladding was zircaloy-4. The nodal configuration consisted of five nodes in the fuel and one node in the cladding. Temperatures were also calculated at the fuel surface, the cladding inside surface, and the cladding outside surface as shown in Figure A-1. The normalized power function shown in Figure A-2 was used for the nuclear rod.

Each of the five nodes within the fuel had the same configuration; the simulation of a typical node, T_3 , is shown in Figure A-3. Integrator A15 integrated the variables of Equation (5) as follows:

$$T_3 = \int \frac{dT_3}{dt} dt = \int \left[\frac{q''' V_3}{V_3 \rho C_3} + \frac{k_{23} A_2}{\ell_{23} V_3 \rho C_3} (T_2 - T_3) - \frac{k_{34} A_3}{\ell_{23} V_3 \rho C_3} (T_2 - T_3) \right] dt. \quad (A-1)$$

The temperature, T_3 , was read into the digital processor through an analog-to-digital converter (ADC 182). The digital processor computed the thermal conductivity, k , and density times heat capacity, ρC_p , as functions of temperature. These variables were sent back on digital-to-analog converters (DAC 83, 84, 85). Multipliers M62 and M64 formed the product of the fuel conductivity, k , and the temperature difference, ΔT , which was then multiplied by the constant geometrical factor $A/\ell V$ of each node set in Potentiometers P15 and Q15.

Multiplier M72 formed the product of power density, q''' , and $kA/\ell V \Delta T$ terms with $1/\rho C_p$ which was the derivative required by Integrator A15 to form the temperature T_3 .

Dimensions (in.)

Rod Outside Diameter	0.4220
Fuel Outside Diameter	0.3660
Cladding Thickness	0.0243
Gap	0.0037

Fig. A-1 Geometry of nuclear rod model.

Fig. A-2 Normalized power generation function for nuclear rod model.

The method used to solve for the fuel surface and cladding surface temperatures was an implicit solution technique (or high speed iteration technique) available on the analog computer. Since no heat storage on the surface was assumed, the heat flux to the surface from an interior node equaled the heat flux lost from the surface. The fuel surface temperature calculation, T_{fs} , in Figure A-4 is included as an example of this technique. The heat transferred to the surface was formed by the output of Multiplier M74 and the geometrical factor A/l was set in Potentiometer Q74. The heat transferred from the fuel surface across the gap was formed in Multiplier M96. Summer A74 maintained the two inputs equal, and the output, T_{fs} , was used to form the necessary temperature differences.

Provision was made for either a variable or constant gap heat transfer coefficient by Function Switch FS-00. Potentiometer P50 provided the constant gap heat transfer coefficient (FS-00 switched right - R), and DAC 90 the variable gap heat transfer coefficient (FS-00 switched left - L).

The latent heat effect of the zircaloy-4 cladding at 1584°F of 19.2 Btu/lb_m was taken into account by switching the net heat flow to the cladding node into a sink represented by Integrator A61 as shown in Figure A-5, until the energy stored by Integrator A61 was equal to the latent heat of the cladding mass. Then the net heat flow was switched back into the cladding node to continue computations of the cladding temperature, T_{clad} . Comparator M5 switched when T_{clad} (A70) equaled 1584°F, and M7 switched when the stored heat in Integrator A61 equaled the latent heat, q .

Also input to the simulation were the film heat transfer coefficient and the water temperature.

Fig. A-3 Circuit for solution of interior temperature T_3 .

ANC-B-774-H

Fig. A-4 Circuit for solution of fuel surface temperature T_{fs} .

ANC-B-776-H

Fig. A-5 Circuit for the latent heat effect of zircaloy cladding.

2. SHEATH TYPE HEATER ROD

The simulation of the sheath type heater rod was identical to that for the nuclear fuel rod except for the heat generation zones and cladding material. The heat generation zone was the stainless steel cladding which was nine mils thick. Both constant and variable gap heat transfer coefficients were used. The outside radius and thickness of the gap at ambient conditions were the same as those of the nuclear fuel rod. The uranium dioxide volume was slightly larger in the sheath type heater rod to account for the difference in cladding thickness of the two rods.

The technique used to determine the heat input to the cladding is described in detail in Appendix C. The required rate of heat input to the cladding of the sheath type heater rod was calculated by maintaining the cladding surface temperature of the nuclear rod constant (determined by steady state film heat transfer coefficient and coolant conditions) during the internal power reduction typical of scram conditions and decay heat generation (Figure A-2) during an LOCA. The rate of heat loss from the nuclear rod was the required rate of heat input to the sheath type heater rod and was independent of the coolant conditions during blowdown.

The implicit solution technique was also used to solve for the electrical power versus time. The circuit is shown in Figure A-6. Programmed on Function Generator DFG-60 was the calculated cladding surface temperature of the nuclear rod for given blowdown conditions. The cladding surface temperature of the nuclear rod was compared with the cladding surface temperature being calculated for the sheath type heater rod in Summer A08. Output of Summer A08 was then fed back into the sheath type heater rod simulation as the required function of electric power versus time; that is, the output maintained the surface temperature of the sheath type heater rod equal to the surface temperature of the nuclear rod. The resulting electric power function was shown in Figure 12 of the main body of this report.

Fig. A-6 Implicit solution technique used to solve for the electrical power generation function required to maintain cladding surface temperature.

3. FILAMENT TYPE ROD

The physical dimensions and properties of the filament type heater rod simulated were typical of those of the heater rods used in the Semiscale Blowdown and Emergency Core Cooling (ECC) Project[A-1] single loop tests. The Nichrome heater was spirally wound and separated from the nickel cladding. The insulation material was boron nitride. The nodal configuration, shown in Figure A-7, consisted of five nodes in the boron nitride and the Nichrome filament, two nodes in the nickel cladding, and one node at the cladding surface. Because of the wide spacing of the Nichrome heater wire, the heat generation zone was assumed to have the thermal properties of the boron nitride insulation. The heat transfer coefficient of the gap was assumed constant at 10,000 Btu/ft²-hr-°F. The thermal properties of the rod materials are included in Appendix B. The same method of solution was used for the heat diffusion equations and surface temperature calculations as previously discussed for the nuclear fuel rod.

The function of electric power versus time has not been analytically determined for the filament type heater rod. The implicit solution technique, as described previously for the sheath type heater rod, was also used for the filament rod.

Fig. A-7 Geometry of filament type heater rod.

ANC-A-778-H

4. COOLANT CHANNEL

The coolant channel simulated was typical of a PWR coolant channel. Two-phase flow was allowed, and saturated steam and saturated water properties were used in the state equations. No subcooled or superheat corrections to the steam and water properties were used in the model. Three inputs for the model were supplied as functions of time: pressure, inlet enthalpy, and mass flux.

The energy equation [Equation (16) of the main body of this report] was rewritten as follows:

$$Mh_c = \int \frac{dMh_c}{dt} dt = \int \left[\dot{m}_{in} h_{in} - \dot{m}_{out} h_{out} + \dot{Q}_{surf} \right] dt. \quad (A-2)$$

Integrator A35 of Figure A-8 formed the Mh_c product for the channel control volume. Outlet enthalpy, h_{out} , was formed in Summer A48, then multiplied by $\dot{m}_{out}$ in Multiplier M36 to form $\dot{m}_{out} h_{out}$. Similarly, h_{in} was multiplied by $\dot{m}_{in}$ in Multiplier M90 to form $\dot{m}_{in} h_{in}$. The heat being transferred from the rod surface, $\dot{Q}_{surf}$, to the coolant was computed in the digital processor.

The mass in the control volume was computed continuously by integration of the difference between the inlet mass flow and outlet mass flow [Equation (14) of the main body of this report]. The inlet mass flow was an input quantity, and the outlet mass flow was computed by the implicit solution technique. The implicit solution technique was based on maintaining the mass times specific volume of the coolant, $M_{cv}v_c$, equal to the volume of the channel. The output of Summer A94, $\dot{m}_{out}$, adjusted the mass computed in Integrator A36 to maintain $M_{cv}v_c$ equal to the volume of the channel. The specific volume, v_c , was based on the thermodynamic state of the coolant:

$$v_c = \begin{cases} v_f, & X \leq 0 \\ v_f + Xv_{fg}, & X > 0 \end{cases} \quad (A-3)$$

The channel enthalpy, h_c , was formed in Divider M92, then combined with h_f and h_g in Divider M94 to form the quality, X , of the channel:

$$X = \frac{h_c - h_f}{h_{fg}} \quad (A-4)$$

Fig. A-8 Coolant model circuit.

5. REFERENCES

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APPENDIX B

COMPUTER SYMBOLISM AND MATERIAL PROPERTIES

APPENDIX B

COMPUTER SYMBOLISM AND MATERIAL PROPERTIES

Table B-I contains a description of the computer symbolism used to describe the hybrid simulation.

Table B-II contains the thermophysical properties used for the rod materials.

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TABLE B-I
COMPUTER SYMBOLS

Potentiometer, $V_0 = (\text{value})_n V_i$

Summer, $V_0 = -[V_1 G_1 + V_2 G_2]$

Integrator, $V_0 = V_{IC} - \int_0^t (V_1 G_1 + V_2 G_2) dt$

Note: Integrator control (●) on master logic unit

High Gain Amplifier, $V_0 = -A(V_1 G_1 + V_2 G_2)$;
 $A \approx 10^8$

Function Generator, $V_0 = f(V_i)$

Multiplier, $V_0 = -\frac{V_1 V_2}{100}$

Divider, $V_0 = 100 \frac{V_2}{V_1}$

Square; $V_0 = -\frac{V_i^2}{100}$

Square Root, $V_0 = 10 \sqrt{|V_i|}$

TABLE B-I (Cont.)

COMPUTER SYMBOLS

Function Switch - Single pole, triple throw

Comparator - Output at one level for $V_1 + V_2 > 0$
 - Output at zero level for $V_1 + V_2 < 0$
 Note: Output on master logic unit

Relay - Switched (+) for $V_1 + V_2 > 0$
 Switched (-) for $V_1 + V_2 < 0$

Track and store - V_0 track V_{in} when logic = one level.
 V_0 stored for logic = zero level.

Note: Logic input on master logic unit

And Gate - Output at one level for all inputs at one level.

Or Gate - Output at one level for any input at one level

TABLE B-II
THERMOPHYSICAL PROPERTY DATA

	Thermal Conductivity $\left(\frac{\text{Btu}}{\text{hr-ft}^2\text{-}^\circ\text{F}}\right)$	Heat Capacity $\left(\frac{\text{Btu}}{\text{lb}_m\text{-}^\circ\text{F}}\right)$	Density at 77°F $\left(\frac{\text{lb}_m}{\text{ft}^3}\right)$	Linear Coefficient of Thermal Expansion, α ($^\circ\text{F}^{-1}$)
Uranium Dioxide [a]	$\frac{3878.1}{T+629.61} + 6.0698 \times 10^{-12}(T+460)^3$ $77 < T < 5072^\circ\text{F}$ Reference B-2	$7.26 \times 10^{-2} + 3.33 \times 10^{-6}T$ $- \frac{4.74 \times 10^{+3}}{(T+460)^2}$ $77 < T < 2240$ $-3.6289 \times 10^{-15}T^4 + 4.6457 \times 10^{-11}T^3$ $-2.0447 \times 10^{-7}T^2 + 3.8303 \times 10^{-4}T - 0.18426$ $2240 < T < 5072^\circ\text{F}$	649.7	$\alpha = 1 \times 10^{-5}; 0 < T < 1000^\circ\text{C}$ $\alpha T = 1.72 \times 10^{-4} + 6.797 \times 10^{-6}T$ $+ 2.896 \times 10^{-9}T^2;$ $T \geq 1000^\circ\text{C}$
Zircaloy-4	$4.14 + 1.044 \times 10^{-2}T$ $-5.27 \times 10^{-6}T^2 + 1.536 \times 10^{-9}T^3$ $572 < T < 2732^\circ\text{F}$ Reference B-3	$6.80 \times 10^{-2} + 1.33 \times 10^{-5}T$ $32 < T < 1376$ 8.6×10^{-2} $T > 1376^\circ\text{F}$ (at 1584°F , a latent heat effect of $19.2 \frac{\text{Btu}}{\text{lb}_m}$ must be added)	404.9	$5.57 \times 10^{-6} + 1.756 \times 10^{-9}T$ $77 < T < 1652^\circ\text{F}$ 9.7×10^{-6} $T > 1652^\circ\text{F}$
Type 304 Stainless Steel	$8.62 + 4.50 \times 10^{-3}T$ $212 < T < 2372^\circ\text{F}$	$0.0919 - 1.505 \times 10^{-5}T$ $+ 3.058 \times 10^{-8}T^2 - 8.36 \times 10^{-12}T^3$ $170 < T < 2000^\circ\text{F}$ Reference B-3	493.0	2.02×10^{-5} $32 < T < 1832^\circ\text{F}$
Boron Nitride	$2.7532 + 1.6738 \times 10^{-2}(T+460)$ $-8.6976 \times 10^{-6}(T+460)^2$ $-1.3217 \times 10^{-9}(T+460)^3$	2.667×10^{-2} $+4.998 \times 10^{-4}(T+460)$ $-1.73 \times 10^{-7}(T+460)^2$ $+2.058 \times 10^{-11}(T+460)^3$	131	
Nickel 200	$40 - 1.756 \times 10^{-2}T$ $0 < T \leq 740^\circ\text{F}$ $16 + 1.486 \times 10^{-2}T$ $740 < T < 2000^\circ\text{F}$ Reference B-4	$1.194 \times 10^{-3}(T-674);$ $T < 674^\circ\text{F}$ $1.025 + 7.62 \times 10^{-4}(T-674);$ $T > 674^\circ\text{F}$	556	
Helium	$3.1264 \times 10^{-3} [T_{\text{avg}} + 460]^{0.5374}$			

[a] The property data are for fuel material having 95% of the theoretical density; theoretical density is $683.9 \frac{\text{lb}_m}{\text{ft}^3}$.
T is in degrees Fahrenheit unless otherwise noted.
Data source is Brassfield et al [B-1] unless otherwise noted.

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APPENDIX C

EVALUATION OF SHEATH TYPE HEATER ROD CONCEPT

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EVALUATION OF SHEATH TYPE HEATER ROD CONCEPT

The concept for sheath type heater rods of the particular design considered was developed by the Battelle, Pacific Northwest Laboratories[C-1]. A method of constructing and heating the electric rods was proposed that would permit simulation of nuclear fuel rods by electric sheath heater rods for LOCA experiments. The construction of the sheath type heater rod was to be the same as that of a nuclear fuel rod, that is, the materials and dimensions were to be the same, but the surface, rather than the interior, of the sheath type heater rod was to be heated.

The required power generation function to the sheath of the heater was determined mathematically so that the surface temperature and heat flux would be the same as those of a fuel rod when the heater was exposed to the same external conditions as the fuel rod and when the thermal properties were constant. The function would give the same surface temperature and the same heat flux to the fluid regardless of the variation of the heat transfer coefficient and coolant conditions during the LOCA. This power generation function is determined and applied as follows:

- (1) The heat flux from the cladding to the coolant as a function of time was determined for the nuclear rod by maintaining the surface temperature constant and equal to that during steady state operation while reducing the power generation rate according to decay heat generation expected during the loss-of-coolant accident.
- (2) The heat flux determined from the cladding to the coolant was input as an electrically generated source at the surface of the sheath heater rod, which was initially at a uniform temperature identical to the cladding surface temperature of the nuclear rod. The electrical heater was then exposed to the LOCA environment by applying the expected effective heat transfer coefficient and sink temperature and determining the actual surface temperature and heat flux of the sheath heater rod.

1. EVALUATION

In order to test mathematically the sheath type heater rod concept, models of the nuclear fuel rod and sheath type heater rod were developed for the hybrid computer. A constant surface temperature was maintained for the nuclear rod, and the heat flux was recorded for a power generation rate expected for LOCA conditions. This heat flux was then input to the sheath type heater rod at the surface of the rod. When the comparison of heat flux and surface temperature was made for the nuclear rod and sheath type heater rod, the results were the same within limits of the recording equipment used.

Since the concept of surface heating and constant thermal properties could not be realized physically, the computer model was changed to include volumetric cladding heating, variable thermal properties, and a variable gap heat transfer coefficient. The results of this model were discussed in the Section III of the main body of this report. A model which

considered volumetric heating, constant thermal properties, and constant gap heat transfer coefficient was also studied using a thickness of three mils for the stainless steel cladding. The results of this run, shown in Figure C-1, for coolant conditions shown in Figures 8 and 9 of the main text, indicate that the time constant due to volumetric heating of a three-mil cladding is small. The physical properties of the rod (gap heat transfer coefficient and thermal properties) are more important parameters than cladding thickness.

Fig. C-1 Cladding surface temperature and heat flux for sheath type rod with three-mil cladding.

2. REFERENCES

- C-1 Program Description: A Preliminary Investigation of Blowdown Heat Transfer, BNWL-CC-2057 (March 1969).