CHANNEL BLOCKAGE ACCIDENT ANALYSIS FOR RESEARCH REACTORS WITH MTR-TYPE FUEL ELEMENTS

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ABSTRACT

It is the purpose of this study to investigate the feasibility of removing the residual decay heat from core of TR-2 ,which is a pool-type research reactor, after a channel blockage accident event and to identify the principal factors involved in cooling process. To analyze this accident scenery, THEAP-I[1] computer code, which is a single phase transient 3-D structure/1-D flow thermal hydraulics code developed with the aim to contribute mainly to the safety analysis of the open pool research reactors, was modified and used. All of the analysis results figured out the fact that the core melting was inevitable in case of an uninterrupted operation (continuous operation) preceding a channel blockage accident of the TR-2 Reactor. Such a result will even be met if the blockage occurs only in a single fuel element. The results of analysis are expressed in terms of temperature field distribution as a function of time.

1. INTRODUCTION

One of the postulated accidents for pool-type research and test reactors is the blockage of the coolant channels which results in loss of flow in the blocked channels. Such an accident would lead to the retention of coolant water in the coolant channels instead of natural or forced circulation of water through the channels. Once the blockage occurs, the flow rate of coolant will be decreased and main heat transfer mechanism in the blocked channels will be conduction that could result in less heat transfer rate when compared to convective heat transfer rate. It is necessary to demonstrate that the residual decay heat can be removed without the hazard of core melting in such a channel blockage accident event.

Nuclear plants are designed to withstand the ground motion caused by the most severe earthquake that is likely to be experienced. From historical records of seismic events in the plant vicinity, the earthquake that would be expected to produce the largest ground motion at the reactor site is predicted. This is called the "safe shutdown earthquake." Analysis must show that the reactor can be tripped and the engineered safety features will function properly if such an earthquake should occur.

In this study the following accident scenario is considered to analyze the partial or entire coolant channel blockage of TR-2 Reactor following a severe earthquake. TR-2 Reactor plant is structurally designed to withstand a maximum horizontal acceleration of 0.4g as the result of the safe shutdown earthquake. It is assumed that the reactor is operating at full power and is essentially instantaneously shut down when the seismic ground motion is detected by the

earthquake detector. It is further assumed that the reactor building will be damaged as a result of the earthquake and some granular materials such as sand, soil will fall down to the open pool surface causing blockage of some coolant channels of the reactor core. The blockage of any channel is considered fully but not partially to simplify the analysis. Otherwise, the heat transfer and conservation equations for blocked channels should be written as for porous media which complicates the modeling too much. In the present modeling, the coolant flow is not permitted by means of natural circulation of water through blocked channels.

From the time when the reactor is shutdown to the time when the coolant channel is blocked by the fallen particles from the roof of the reactor building, the removal of decay heat from the reactor core is considered by means of natural convection of coolant water. The time at which the channels become blocked is considered the "start" of the Channel Blockage Accident, i.e., t=0 in the computer modeling used in the analysis. On the other hand, after the blockage, main decay heat removal mechanism of the core is considered as conduction in the blocked channels and natural convection in the unblocked channels. Conduction and convection to the neighboring elements is also considered in the modeling.

2. PROBLEM DEFINITION

2.1. Problem Geometry

The problem general geometry is illustrated in Figure-1.a. A set of thermally interacting heated/coolant channels are kept together through one support plate located at a elevation. The whole bundle and the support plate are totally immersed into water. The thermal interaction among channels and the support plate occur through small gaps of finite (or zero) width. The channel structure is cooled through an internal flow stream naturally or forcibly.

2.2. The Conservation Equations

The equations to be solved are the following ;

- The channel structure energy equation to produce the structure temperature field.
- The support plate energy equation to produce the support plate temperature.
- The internal stream conservation equations (mass + momentum + energy).

2.2.1. The Channel Structure Energy Equation

The transient, 3-D energy equation in Cartesian coordinates (x,y,z) at a certain elevation for a particular channel (i,j) is given by the expression;

$$\rho c_p \gamma_v A_z dz \frac{dT_{w,i,j}}{dt} - \gamma_v A_z dzq''' = k\gamma_z A_z \frac{\partial^2 T_{w,i,j}}{\partial z^2} dz - dzdyh(T_{w,i,j} - T_{w,i+1,j}) - dzdyh(T_{w,i,j} - T_{w,i-1,j}) - dzdxh(T_{w,i,j} - T_{w,i,j+1}) - dzdxh(T_{w,i,j} - T_{w,i,j-1}) - h_f P_{wf} dz(T_{w,i,j} - T_{f,i,j}) \quad (1)$$

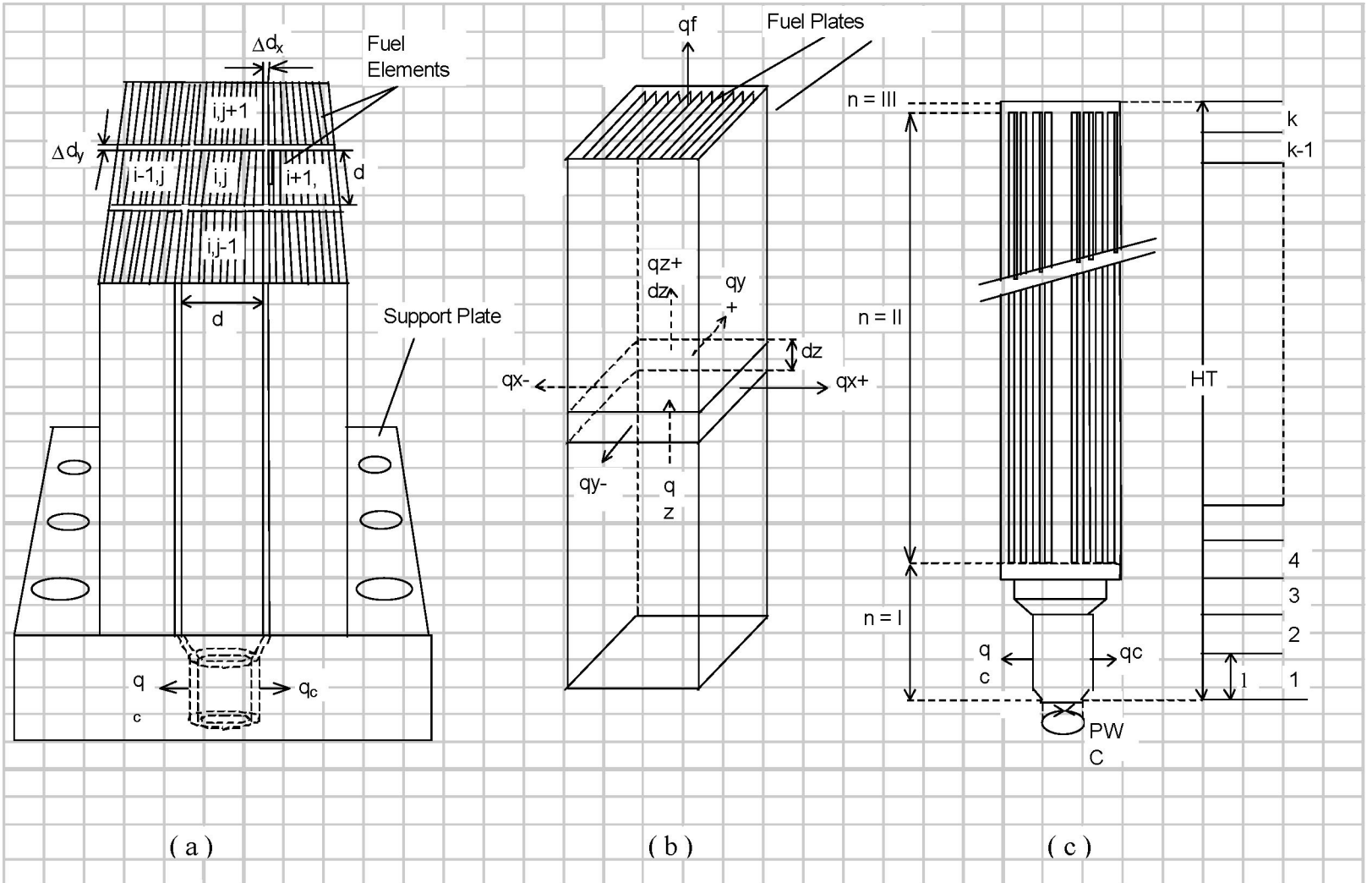


Figure-1 . Thermal Interaction Between Fuel Elements and Support Plate

In the above equation the concept of volume and surface porosity has been adopted which is an efficient tool for thermal hydraulic analysis in complicated geometries [2].

It is necessary to remind that volume porosity expresses the percentage of solid material within a certain volume whereas surface porosity expresses the percentage of solid material in a certain surface.

2.2.2. The Support Plate Heat Balance Equation

Since the support plate temperature is considered uniform the heat balance equation is written as

$$(\rho c_p \gamma_v A_z)_{i,j} \frac{dT_{w,i,j}}{dt} - (k\gamma_z A_z)_{i,j} \frac{\partial^2 T_{w,i,j}}{\partial z^2} + (h_c P_{wc})_{pl}(T_{w,i,j} - T_{pl}) = 0 \qquad (2)$$

where

T_w : Temperature of bottom part of fuel

T_{pl} : Temperature of support plate

P_{wc} : Wetted perimeter of bottom part of fuel

2.2.3. The Internal Stream Conservation Equations

One dimensional flow is considered to be sufficient. The **continuity equation** will be approximated as

$$\dot{m}_{fl}(z,t) = \dot{m}_{fl}(t) \qquad (3)$$

The flow rate will be calculated from the **integrated momentum equation** which is given by the relation

$$\sum_k \left[\frac{\partial(\dot{m}_{fl})}{\partial t} \frac{H_l}{A_{fl}} + \frac{\dot{m}_{fl}^2}{A_{fl}^2}\left(\frac{1}{\rho_l} - \frac{1}{\rho_0}\right)_l + c_l \frac{\dot{m}_{fl}}{2D_H^2 A_{fl}} \int_0^l \frac{\mu_{fl}}{\rho} dz \right]$$
$$+ \sum J_n \frac{\dot{m}_{fl}^2}{2\rho A_{fl}^2} + \sum_k \int_0^l \rho g dz = \sum_k (P_0 - P_l) \qquad (4)$$

where " k " is the straight sublengths within the fuel element.

The **energy equation** is expressed as

$$(\rho_{fl} c_p A_{fl})_{i,j} \frac{\partial T_{f,i,j}}{\partial t} + (\dot{m}_{fl} c_p)_{i,j} \frac{\partial T_{f,i,j}}{\partial z} = (h_{fl} P_{wf})_{i,j} (T_{w,i,j} - T_{f,i,j}) \quad (5)$$

2.3. Numerical Methods of THEAP-I Code

The above mentioned energy equations are transformed into finite difference ones using using standard finite difference techniques. The resulting finite difference equations are nonlinear and fully implicit with respect to time. They can be solved either by the Gauss point iteration method, or by line iteration method [3].

2.4. The Residual Heat

The cause for temperature increase in the core following the accident is residual heat. For an irradiated core of initial power P_o and continuous total irradiation time T, the power level at time t after shutdown, it is approximated by familiar Way-Wingler formula.

$$q(t) = 0.0622 P_0 (t^{-0.2} - (3600 * T + t)^{-0.2}) \quad (6)$$

However, in many cases like in TR-2, does not work on a 24-hours basis, but on a daily schedule. Other factors that are expected to affect equivalent irradiation time are, weekly schedule, the number of fuel cycles, the shutdown time for fuel shuffling etc.

Let us assume that the reactor is working on a **HRSPD** hours/day basis, **NPDAW** days/week. Each cycle lasts for **ld** days in which the last NDSUF days are required to perform the necessary fuel shuffling for the next cycle. Each fuel element has performed **nc** cycles. The power of the element at the n-th (n = 1,2,3,... ,nc) cycle is Q_{0n} .

The heat source is given by the relation

$$Q_F = \sum_{1}^{nc} f_n Q_{0n} \quad (7)$$

$$f_n = 0.0622 \sum_{l=1}^{l=ld} (t_l^{-0.2} - (3600 * HRSPD + t_l)^{-0.2}) \quad (8)$$

$$t_{ln} = (ld - NDSUF) * (n-1) - INT \frac{ld * (n-1)}{7}$$
$$* (7 - NDAYPW) + INT \frac{(l-1)}{7} * (7 - NDAYPW) \quad (9)$$
$$* HRSPD * 3600 + t$$

2.5. Heat Transfer Coefficients

2.5.1. Heat Transfer From the Outer Surfaces to the Environment

Heat is transferred from the outer surfaces to the environment mainly through natural convection. Thus the corresponding heat transfer coefficient can be expressed as

$$h = \begin{cases} C_L*(k/L)*(PR*GR_L)^{1/4}; GR_L \langle GR_L{}^* \\ C_T*(k/L)*(PR*GR_L)^{1/3}; \text{diğer} \end{cases} \qquad (10)$$

$GR_L = \rho^2 g\beta(T_W - T_r)*L^3/\mu^2$: Grashof Number

$PR = \mu c_p / k$: Prandtl Number

L : Appropriate length scale

T_r : Reference fluid temperature

C_L, C_T : constants

GR_L : is a representative Grashof Number values, indicating transmission from laminar to turbulent flow.

2.5.2. Interelement Heat Transfer

The interelement gaps in the open pool research reactors are usually rather small (order of 1 mm). Therefore, it seems a good approximation to assume that the main heat transfer mechanism is conduction. The equivalent heat transfer coefficient can be expressed as

$$h = k / d \qquad (11)$$

2.5.3. Internal Fuel element Heat Transfer Coefficient

The major heat transfer mechanism in this case is considered to be natural convection throughout the channels which are not blocked in the accident. And the heat transfer coefficient for these channels is given as in equation (10). For the channels blocked in the accident heat is transmitted via conduction and the equivalent heat transfer coefficient is given as in equation (11).

3.TR-2 REACTOR

The calculations has been carried out for the 12-84 fuel cycle, and the configuration for this core is shown in Figure-2.

			Water Box	Water Box	Water Box		
5	^{72}	Be ^{73}	S-110^{74}	S-118^{75}	S-112^{76}	Be^{77}	^{78}
4	Be^{62}	S-109^{63}	C-017^{64}	S-108^{65}	C-015^{66}	S-104^{67}	Be^{68}
3	Be^{52}	S-114^{53}	S-101^{54}	I-02^{55}	S-111^{56}	S-103^{57}	Be^{68}
2	Be^{42}	S-102^{43}	C-016^{44}	S-107^{45}	C-012^{46}	S-113^{47}	Be^{68}
J=1	^{32}	Be^{33}	S-106^{34}	S-116^{35}	S-115^{36}	Be^{37}	^{38}
			Water Box	Water Box	Water Box		

i = 1 2 3 4 5

S: Standard Fuel Element

C: Control Fuel Element

Be: Berilium Block

I : Irradiation Fuel Element

Figure-2. 12-84 Fuel Cycle Core Configuration

Properties for fuel elements forming the 12-84 cycle core is given in Table-1.

Table-1. Properties of fuel elements forming 12-84 cycle core

	Standard	Control	Irradiation
Enrichment	%93	%93	%93
U-235 g/plate	12.2	12.2	12.2
Number of fuel plates	23	17	12
U-235 g/element	280.6	207.4	146.4
Thickness of fuel plate (mm)	1.27	1.27	1.27
Active fuel thickness (mm)	0.51	0.51	0.51
Clad thickness (mm)	0.38	0.38	0.38
Plate width (mm)	71.0	71.0	71.0
Active fuel width (mm)	59.2-65.4	59.2-65.4	59.2-65.4
Inner plate length (mm)	625.9	625.9	625.9
Outer plate length (mm)	709.0	1255.0	709.0
Active fuel length (mm)	586-610	586-610	586-610
Gap betwwen plates (mm)	2.1	2.1	2.1
Fuel box dimensions (mm x mm)	76.1 x 80	76.1 x 80	76.1 x 80
Fuel element length (mm)	873	1419	873
Fuel element mass (g)	5600	7800	4550
Mass of active fuel in a fuel plate (g)	67.7	67.7	67.7
Weight fraction of active fuel in a fuel elemet (W_m)	0.278	0.1535	0.1675

4. CALCULATIONS

To analyze the a hypothetical accident discussed before, it was decided to use THEAP-I computer code, a single phase transient 3-D structure/1-D flow thermal hydraulics code developed with the aim to contribute mainly to the safety analysis of the open pool research reactors.

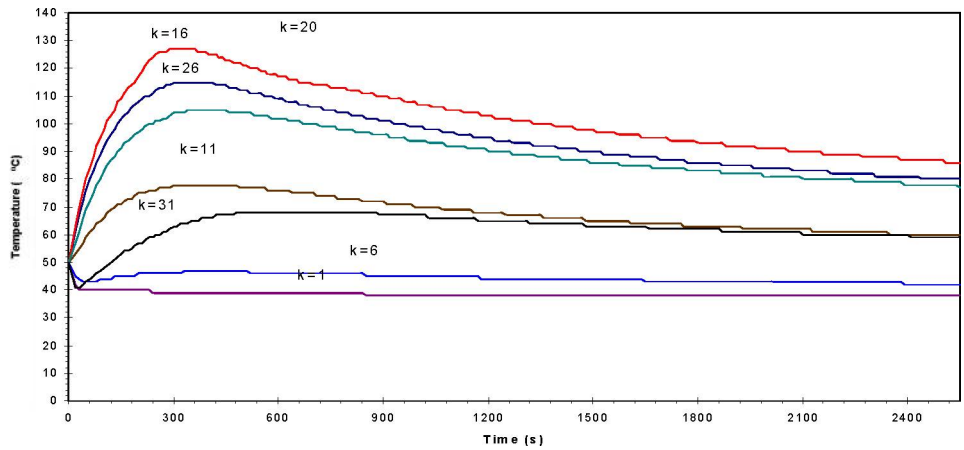
At the first stage of the problem modeling, THEAP-I was modified to correspond the current accident scenario. Initially, the blocked channels are defined by means of setting flow rates to "zero" for the selected channels in the subroutine of THEAP-I which calculates the natural flow rates (NATFLO). To introduce the blocked channels to the code a specific hydraulics diameter value is assigned for blocked channels in the problem input data so that whenever hydraulics diameter matches to assigned value, calculations will be done for the blocked channels. Then, to evaluate the equivalent heat transfer coefficients for the blocked channels the convection term is dismissed in the subroutine (SCOOLD) calculating equivalent heat transfer coefficients for vertical channels [4].

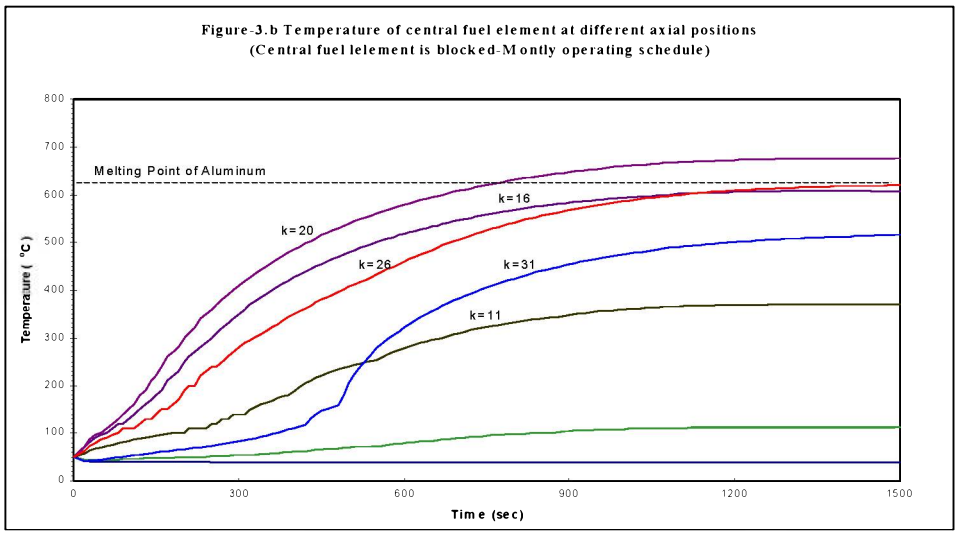
The second step was to complete the physical modeling by modifications in the code in order to calculate boiling heat transfer coefficients for the channel sub-regions where the boiling temperature is reached . For THEAP-I is a single phase thermal hydraulics code a rigorous approximation, which is more conservative than the boiling itself, was done to simulate the boiling, To do this the bubbles in the boiling regions are treated as air. Because of the worse heat transfer coefficients of air than vapor at the same temperature this assumption leads to higher temperature estimations and in view of safety aspects is more conservative than the actual situation.

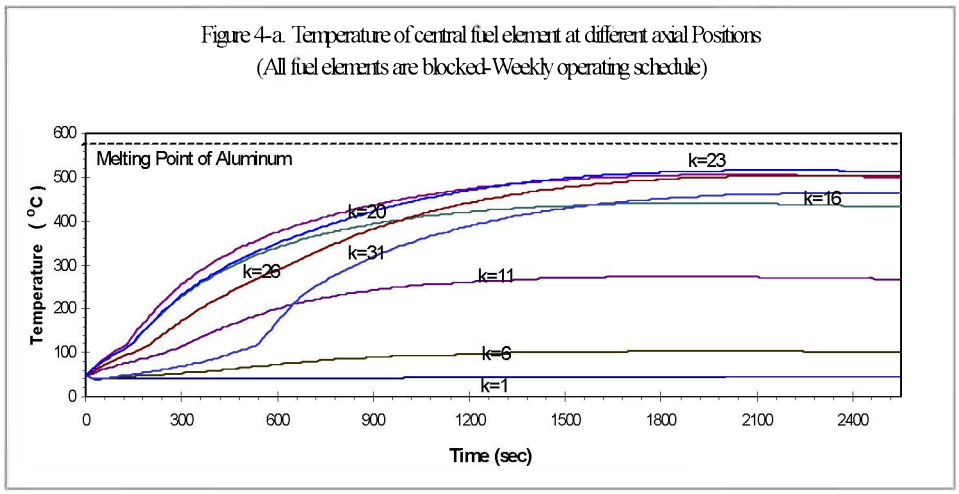
4.1. Blockage of Central Fuel Element

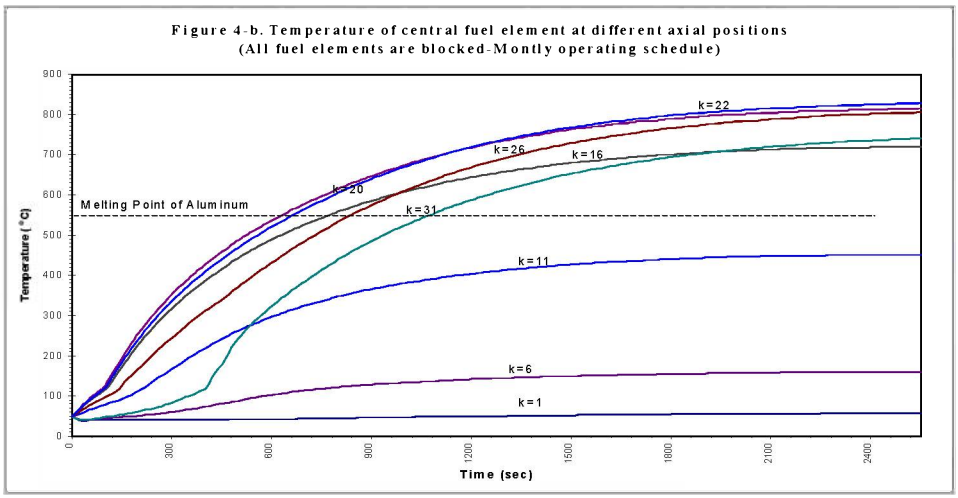
Following the code modifications, centered fuel element(I-02) of TR-2 Reactor was considered blocked for the first run of the code. Since the reactor's operating schedule has an essential role in the calculation of the core decay heat two operating schedules was considered :

Figure-3.a. Temperature of central fuel element at different axial locations
(Central fuel element is blocked -Weekly operating schedule)




Figure-3.b Temperature of central fuel element at different axial positions
(Central fuel lelement is blocked-Montly operating schedule)
Melting Point of Aluminum
k=20
k=16
k=26
k=31
k=11
Temperature (°C)
Time (sec)


Figure 4-a. Temperature of central fuel element at different axial Positions
(All fuel elements are blocked-Weekly operating schedule)
Melting Point of Aluminum
k=23
k=16
k=20
k=31
k=26
k=11
k=6
k=1
Temperature (°C)
Time (sec)


Figure 4-b. Temperature of central fuel element at different axial positions
(All fuel elements are blocked-Montly operating schedule)
Melting Point of Aluminum
k=22
k=26
k=16
k=20
k=31
k=11
k=6
k=1
Temperature (°C)
Time (sec)

1. weekly operating schedule (operation of five day per week, six hours a day), 2. one month continuous operation. The results of the analysis for each operating schedules are given in terms of temperature field distribution of the central fuel element as a function of the time. The results are presented graphically in Figure-3.a-b.

4.2. The Blockage of All Fuel Elements

The worst case, blockage of the whole core channels, was considered for both operating schedules too. The results of analysis for each schedule are given in terms of temperature fields distribution of the centered fuel element as a function of the time. The results are presented graphically in Figure-4.a-b.

5. CONCLUSION

All of the analysis results figured out the fact that the core melting was inevitable in case of an uninterrupted operation (continuous operation) preceding a channel blockage accident of the TR-2 Reactor. Such a result will even be met if the blockage occurs only in a single fuel element. But this is not the case for the reactor's routine operating schedule. The reactor's operation scheme had been weekly operating schedule until it was ceased in the mid-summer of 1995 in order to re-evaluate safety aspects of it. The analysis also shows that core melting will not occur if the reactor is operated in weekly schedule which is not a continuous operating schedule that yields less decay heat rate compared to continuous operation before the accident. As a result, we could easily say that in routine operation (weekly operation) core melting will not occur following a channel blockage accident caused by an earthquake even if all coolant channels are blocked.

It should also be noted that one of the important program input parameters to be determined was the time "zero", the time passed after earthquake until blockage of the channels occurred. Because, selection of different "zero" times will result in different initial decay heat and initial temperature fields of the core elements at the time of blockage. Successive runs showed that the behavior of the thermal hydraulics system was purely sensitive to the input parameter "zero" time in case, it is in the orders of up to a few minutes. So, a reasonable value of 10 seconds assigned for "zero" time in all of the above four runs.

The results figured out the fact that a detailed channel blockage accident analysis considering two phase heat transfer mechanism must be made if TR-2 Reactor is planned to be operated in a continuous operating program. The reevaluation of the accident would be necessary because of the higher temperature estimations of the current study. Otherwise, the current results indicate the risk of core meltdown in case of a channel blockage accident.

Further amendments could be made by taking into consideration of boiling heat transfer coefficients in the coolant channel regions where boiling occurs.

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