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POTENTIAL INFLUENCE OF CROSSFLOW AND
RADIATION HEAT TRANSFER ON LOFT LOCA BEHAVIOR

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SUMMARY

The MOXY/SCORE computer program [1] has been used to evaluate the influence of crossflow and radiation heat transfer on LOFT cladding temperatures during the blowdown phase of L2-4.

The results of this study indicate that the radiation heat flux is responsible for 100 K reduction of peak cladding temperature during blowdown. They also indicate that, if the boundary conditions are specified uniformly over upper and lower plenums, the crossflow will be responsible for another 25 K reduction of peak cladding temperature during blowdown.

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ABSTRACT

The MOXY/SCORE computer program^[1] is used to evaluate the influence of crossflow and radiation heat transfer on LOFT cladding temperatures during the blowdown phase of L2-4.

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1.0 INTRODUCTION

The thermal-hydraulic behavior of the LOFT reactor during a planned LOCE has been investigated mostly by one-dimensional (1-D) analytical codes. The predictions of 1-D codes would be unrealistic if there is a strong influence of three-dimensional (3-D) effects. In particular, without taking 3-D effects such as coolant crossflow and radiation heat transfer into account, a 1-D code could predict an unrealistically high peak clad temperature.

The radial power gradient across the LOFT core is much greater than that of a large commercial PWR. As can be seen from Figure 1.0.1, the fuel rods of high power density are concentrated in the central region. Contrarily, a commercial PWR core is characterized by a fairly flat radial power distribution. As a result of its larger radial power gradient, it is reasonable to expect that other thermal-hydraulic parameters of the LOFT core, e.g., cladding temperature and coolant quality, will also have larger radial gradients compared with those of a large PWR core. The implication is that there could be a stronger influence of 3-D effects on LOFT core LOCA behavior than expected for a large PWR core.

As indicated previously, a 1-D code might provide more conservative prediction of peak clad temperature if there is a stronger influence of 3-D effects. Therefore, a 1-D code LOCA behavior prediction for the LOFT reactor is likely to be more conservative than that would be the case for a large PWR. Thus, the degree of conservatism of a 1-D code established by the comparison between its predictions and LOFT test results may not be a true indication of the conservatism of that code in predicting the LOCA behavior of a large PWR.

Accordingly, a preliminary study of the potential influence of 3-D effects on LOFT reactor LOCA behavior was carried out. The objectives of this initial study, specifically, are:

- 1) Top Numbers - Average Pin-Power/Average Core Power
- 2) Bottom Numbers - Average Power in Fuel Pin at the Hot Plane/Average Core Power

Therefore Item (1) gives Radial-Peaking Factors and the Ratio of Item (2) to Item (1) in any Pin gives the Axial-Peaking Factor for that Pin.

[illegible][illegible]

FIG. 1.0.1: 200 EFPH POWER DISTRIBUTION
FOR 1/8 LOFT CORE

- (1) To evaluate the potential influence of crossflow and radiation heat transfer on LOFT reactor LOCA behavior during blowdown. The peak clad temperature, due to its pivotal role on reactor safety, is the parameter of greatest interest.
- (2) To verify and improve, where possible, the 3-D computer program MOXY/SCORE^[1]. MOXY/SCORE is the key tool used in this study. This program, due to its relatively long running time, must be further improved before being used on a more frequent basis.

This document describes the results of this study. The remainder of the report is organized into the following sections: Section 2 - Analysis Approach, Section 3 - Computational Procedure, Section 4 - Results and Analyses, and Section 5 - Conclusions and Recommendations.

2.0 ANALYSIS APPROACH

The LOFT L2-4 test^[2] (200% double-ended offset shear cold leg break with maximum linear heat generation rate = 52.5 kW/m) was selected as the reference case for LOCA analysis because it will produce the most severe core thermal-hydraulic response among the L2 series of tests.

The analytical approach of this study is dictated by the basic goal of quantifying the individual effects of crossflow and radiation heat transfer. The basic approach can be briefly described by:

- (1) Using MOXY/SCORE, carry out a 3-D core-wide calculation with the radiation heat transfer mechanism active (view factors between heat slabs are important input for radiation heat transfer calculation. The details of view factor calculation are given in Appendix A). For a 3-D calculation, each symmetric quarter of the reactor core is divided into 16 flow channels and 15 axial levels (see Figures 2.0.1 and 2.0.2).
- (2) Using MOXY/SCORE, carry out a 3-D core-wide calculation with the radiation heat transfer mechanism turned off.
- (3) Using MOXY/SCORE, carry out a 1-D calculation along the hot channel (Channel #1 in Figure 2.0.1).
- (4) Using MOXY/SCORE, carry out a 1-D calculation along a peripheral channel (Channel #8 in Figure 2.0.1).
- (5) The differentials of the results of these calculations are used to evaluate the individual effects of crossflow and radiation heat transfer on hot channel and peripheral channel LOCA behavior.

FIG. 2.0.1. LOFT CORE CROSS-SECTIONAL VIEW SHOWING
16 FLOW CHANNELS FOR MOXY/SCORE MODEL

K = FORM LOSS COEFFICIENT

FIG. 2.0.2. FUEL ROD ASSEMBLY AXIAL CELL ARRANGEMENT
FOR MOXY/SCORE MODEL

The MOXY/SCORE calculations are based on a number of assumptions. They are:

- (1) Gap conductance is time-independent. This assumption is made to simplify the calculations and reduce computer running time. The results of the individual calculation could have been affected greatly by this assumption. However, the differentials of the calculational results are unlikely to be sensitive to this assumption.
- (2) The specified boundary conditions (inlet axial velocity, inlet internal energy, outlet internal energy and outlet pressure) are uniform over the core inlet and outlet. This assumption is a result of the lack of knowledge of the spatial distribution of these boundary conditions during a transient loss-of-coolant experiment.
- (3) The guide tubes are not included in the heat transfer calculation. The surface temperature of these tubes is much lower than that of fuel rods. Taking them into account in the heat transfer calculation will amplify the effect of radiation heat transfer and further reduce the peak clad temperature.
- (4) Because radiation heat exchange between heat slabs of different axial levels is small, only radiation heat transfer among heat slabs of the same axial level will be considered.

3.0. CALCULATIONAL PROCEDURE

A typical MOXY/SCORE calculation includes two major steps, i.e., (1) Steady State Initiation and (2) Transient Calculation.

3.1 Steady State Initiation

A RELAP4^[3] system calculation provided the initial outlet pressure, initial inlet internal energy, initial inlet quality and initial inlet axial velocity. Based on these initial boundary conditions and prescribed initial core power distribution, an explicit 1-D MOXY/SCORE calculation (in this calculation, each individual flow channel is considered independently of each other, with initial crossflow being zero) will give an approximate steady state thermal-hydraulic solution.

3.2 Transient Calculation

The steady state thermal-hydraulic solution calculated by an explicit 1-D MOXY/SCORE calculation is a crude solution and must be refined. To obtain a more exact steady state solution, a MOXY/SCORE transient calculation with constant fluid boundary conditions and core power distribution is always initiated for 0.1725 sec (350 time-steps) prior to the beginning of blowdown. At the end of this initial calculation, core thermal-hydraulic conditions have approached closer to a steady state solution with initial crossflow (it is observed that the amplitudes of pressure oscillations have been reduced by a factor of 10^4 by this process).

Blowdown calculation starts at 0.1725 sec after the beginning of the transient calculation. The fluid boundary conditions during the blowdown period are also provided by a RELAP/4 system calculation. The end time of the transient calculation is 19.5 sec. This end time is chosen because:

- (1) The two-phase model of MOXY/SCORE is the HEM (Homogeneous Equilibrium Model). Core thermal-hydraulic conditions are neither homogeneous nor equilibrium after $t = 19.5$ sec.
- (2) Fluid velocities increase sharply after $t = 19.5$ sec. In order to maintain numerical stability, the time-step size must also be reduced sharply, greatly increasing the expense of the calculation.

In the process of carrying out MOXY/SCORE calculations, some numerical problems were encountered. These problems and their solutions will be described in Appendix B.

4.0 RESULTS AND ANALYSES

As indicated in Section 2, four MOXY/SCORE calculations form the basis to quantify the effects of radiation heat transfer and crossflow on peak cladding temperature. These calculations can be summarized briefly as:

- (1) 3-D calculation with the radiation heat transfer mechanism turned on.
- (2) 3-D calculation with the radiation heat transfer mechanism turned off.
- (3) 1-D hot channel (Channel #1 in Figure 2.0.1) calculation.
- (4) 1-D peripheral channel (Channel #8 in Figure 2.0.1) calculation.

These calculations, from now on, will be designated as Run #1, Run #2, Run #3, and Run #4, respectively. The results of these calculations are presented and evaluated in this section.

To facilitate the presentation and evaluation of calculational results, more background information about MOXY/SCORE calculations will be provided as follows:

- (a) The reactor core consists of four symmetric quarters. As shown in Figures 2.0.1 and 2.0.2, each quarter is divided into 16 flow channels and 15 axial levels. Considering the diagonal symmetry of this quarter core, it can be seen that there are 150 independent fluid cells.

- (b) Two heat slabs (canister and fuel rod heat slabs) are associated with each fluid cell in Channels #4, #8, #12, #15, #14 and #13 of Figure 2.0.1. One heat slab (fuel rod heat slab) is associated with each fluid cell in Channels #1, #2, #3, #5, #6, #7, #9, #10, and #11 of Figure 2.0.1. There is no heat slab associated with Channel #16. Considering the diagonal symmetry, it can be seen that there are 180 independent heat slabs.
- (c) The Fluid Cell (FC) which is located at n-th axial level in channel #m will be designated as FC m-n. Similarly, the corresponding Fuel Rod Heat Slab (FRHS) and Canister Heat Slab (CHS) will be designated as FRHS m-n and CHS m-n, respectively.
- (d) Because there are so many independent fluid cells and heat slabs, the complete description of the MOXY/SCORE calculational results will not be presented. Only the calculational results for the axial levels of peak cladding temperature for both Channel #1 (central channel) and Channel #8 (a peripheral channel) will be presented. The axial levels of peak cladding temperature in Channel #1 and Channel #8 as indicated by MOXY/SCORE calculations are listed in Table 4.0.1.

TABLE 4.0.1

AXIAL LEVELS OF PEAK CLADDING TEMPERATURE

	Axial Level of Peak Cladding Temperature			
	Run #1	Run #2	Run #3	Run #4
Channel #1 (central channel)	9-th	9-th	8-th	
Channel #8 (A peripheral channel)	8-th	8-th		8-th

- (e) Because each fluid cell in Channel #1 contains only one heat slab (see Part (b)) and it is assumed that radiation heat transfer does not occur between heat slabs of different axial levels, Run #3 (which is a 1-D calculation along Channel #1) involves no radiation heat transfer calculation. Contrarily, each fluid cell in Channel #8 contains two heat slabs and there is a radiation heat transfer calculation in Run #4 (a 1-D calculation along Channel #8).
- (f) There is no crossflow for any 1-D calculation (Run #3 or Run #4).
- (g) Convective heat transfer modes which are assumed to apply in the core during blowdown are:
 - Mode 1 - Subcooled Liquid Forced Convection
 - Mode 2 - Nucleate Boiling
 - Mode 3 - Forced Convective Vaporization
 - Mode 4 - Transient Boiling
 - Mode 5 - Stable Film Boiling
 - Mode 6 - Low Flow, Low Void Fraction Film Boiling (modified Bromley)
 - Mode 7 - Low Flow, High Void Fraction Film Boiling (free convection and radiation)
 - Mode 8 - Superheated Vapor Forced Convection.
- (h) Relationship between convective and radiative heat transfer to fluid is discussed in Appendix C.

The presentation and evaluation of MOXY/SCORE calculational results will be given in the following subsections: Subsection 1 - Fluid Velocities, Subsection 2 - Fluid Thermodynamics Properties, Subsection 3 - Fuel Rod Heat Transfer Parameters, and Subsection 4 - Fuel Rod Temperature Distribution. In these subsections, the abbreviations established in Part (c) of background information will be used extensively.

It is again emphasized that, because several important assumptions have been made to simplify the calculations, the analytical results of each individual calculation do not represent the predictions of actual core thermal-hydraulic response under LOCA conditions.

4.1 Fluid Velocities

The histories of a set of fluid velocities are shown in Figures 4.1.1 through 4.1.6 (x-direction is the direction of axial flow; y and z directions are shown in Figure 2.0.1). The major features of these figures are:

- (a) Crossflow (flow in y or z direction) is negligible compared with axial flow in both FC 1-9 (Fluid Cell at the 9-th axial level in Channel #1) and FC 8-8. A review of crossflow in other fluid cells also supports this trend. Except at the core outlet, crossflow generally is very small. The implication of this trend is that a strong influence of crossflow on core thermal-hydraulic behavior is not expected during blowdown.
- (b) There is a strong influence of radiation heat transfer on x-velocity in FC 1-9 (the fluid cell at 9-th axial level in Channel #1) after $t = 12$ sec. (t = time after the start of transient calculation). This phenomenon is a result of the fact that radiation heat flux becomes comparable with convective heat flux in FC 1-9 after $t = 12$ sec (see Subsection 4.3).

Contrarily, the influence of radiation heat transfer on x-velocity in FC 8-8 is rather insignificant. FC 8-8 is a peripheral fluid cell and radiation heat flux is only a small portion of total heat flux during blowdown (see Subsection 4.3).

FIG. 4.1.1. X-VELOCITIES IN CHANNEL #1

X-VELOCITY (METER/SECOND)

FIG. 4.1.2. X-VELOCITIES IN CHANNEL # 8

FIG. 4.1.3. Y-VELOCITIES IN CHANNEL # 1

17

FIG. 4.1.4. Y-VELOCITIES IN CHANNEL # 8

FIG. 4.1.5. Z-VELOCITIES IN CHANNEL # 1

FIG. 4.1.6. Z-VELOCITIES IN CHANNEL # 8

4.2 Fluid Thermodynamic Properties

The histories of several thermodynamic properties (quality, density, temperature and pressure) are shown in Figures 4.2.1 through 4.2.8. (It is noted that the histories of mixture densities, instead of fluid densities, are shown in Figures 4.2.3 and 4.2.4. However, mixture density and fluid density are related by a simple relation, i.e., mixture density = fluid density x (fraction of channel area occupied by fluid). For FC 1-9, FC 1-8 and FC 8-8, mixture density = fluid density x 0.5682. The major features of these figures include:

- (a) Quality tends to go up and density tends to go down when flow reverses (see Figures 4.1.1 and 4.1.2). This phenomenon is expected because high quality fluid is drawn from the upper plenum when flow reverses.
- (b) Temperature-time and pressure-time curves generally are fairly smooth during most of the blowdown period. Near the beginning and the end of blowdown, some sharp turns do occur on temperature-time curve. These sharp-turns, as can be seen from Figures 4.1.1, 4.1.2, 4.3.1 and 4.3.2 are the results of dryout and fluid velocity changes. Temperature-time curves also indicate that radiation heat transfer can cause an earlier superheat.

4.3 Fuel Rod Heat Transfer Parameters

This subsection will focus on the following heat transfer parameters:

- (1) Cladding Surface Convective Heat Transfer Modes
- (2) Cladding Surface Convective Heat Transfer Coefficient
- (3) Cladding Surface Convective Heat Flux
- (4) Cladding Surface Radiation Heat Flux
- (5) Cladding Surface Metal-Water-Reaction Heat Flux.

FIG. 4.2.1. FLUID QUALITIES IN CHANNEL # 1

FIG. 4.2.2: FLUID QUALITIES IN CHANNEL # 8

FIG. 4.2.3. FLUID DENSITIES IN CHANNEL # 1

FIG. 4.2.4. FLUID DENSITIES IN CHANNEL # 8

FLUID TEMPERATURE (K)

FIG. 4.2.5. FLUID TEMPERATURES IN CHANNEL # 1

FLUID TEMPERATURE (K)

FIG. 4.2.6. FLUID TEMPERATURES IN CHANNEL # 8

FIG. 4.2.7. FLUID PRESSURES IN CHANNEL # 1

FIG. 4.2.8. FLUID PRESSURES IN CHANNEL # 8

The histories of these parameters for certain FRHS's (Fuel Rod Heat Slabs) are shown in Figures 4.3.1 through 4.3.10. From these figures, one observes that:

- (a) Convective heat transfer is dominated by film boiling during most of the blowdown period. For FRHS 1-9 (calculated by Run #1 and Run #2) and FRHS 1-8 (calculated by Run #3), film boiling occurs approximately between $t = 1$ sec and $t = 17$ sec. For FRHS 8-8 (calculated by Run #1, Run #2 and Run #4), film boiling occurs approximately between $t = 3$ sec and $t = 17$ sec.
- (b) Following the incipience of film boiling, fuel rod cladding temperatures generally increase sharply (see Figures 4.4.7 and 4.4.8). As a result, radiation heat flux also increases sharply. Run #1 indicates that, for FRHS 1-9, radiation heat flux increases rapidly following the incipience of film boiling and becomes comparable with convective heat flux (both are in the range of 50000 watt/m^2) after $t = 12$ sec. For FRHS 8-8, both Run #1 and Run #4 indicate that radiation heat flux also increases sharply following the incipience of film boiling. For some intervals after $t = 12$ sec, radiation heat flux may become an appreciable portion of total heat flux.
- (c) The radiation heat flux of FRHS 8-8 calculated by Run #1 is about 20% less than that calculated by Run #4 (see Figure 4.3.8). This difference may be explained by the following argument: Run #4 is a 1-D calculation and radiation heat transfer can only occur between one fuel rod heat slab and one canister heat slab in the same fluid cell. Contrarily, Run #1 is a 3-D calculation and radiation heat transfer can occur among all the heat slabs in the same axial level. Because FRHS 8-8 has lower cladding temperature compared with most of FRHS's in the same axial level, the net effect of radiation heat transfer

FIG. 4.3.1 CONVECTIVE HEAT TRANSFER MODES IN CHANNEL # 1

(HEAT TRANSFER MODES ARE DEFINED ON PP. 12)

CONVECTIVE HEAT TRANSFER MODE

FIG. 4.3.2 CONVECTIVE HEAT TRANSFER MODES IN CHANNEL # 8

(HEAT TRANSFER MODES ARE DEFINED ON PP.12)

CONVECTIVE HEAT TRANSFER COEFF. (WATT/M²-K)

FIG. 4.3.3. CONVECTIVE HEAT TRANSFER COEFFICIENTS IN CHANNEL # 1

CONVECTIVE HEAT TRANSFER COEFF. (WATT/M²-K)

FIG. 4.3.4. CONVECTIVE HEAT TRANSFER COEFFICIENTS IN CHANNEL # 8

FIG. 4.3.5. CONVECTIVE HEAT FLUXES IN CHANNEL # 1

CONVECTIVE HEAT FLUX (WATT/M²) FIG. 4.3.6. CONVECTIVE HEAT FLUXES IN CHANNEL # 8

FIG. 4.3.7. RADIATION HEAT FLUX IN CHANNEL #1

FIG. 4.3.8. RADIATION HEAT FLUXES IN CHANNEL # 8

METAL-WATER-REACTION HEAT FLUX (WATT/M²)

FIG. 4.3.9. METAL-WATER-REACTION HEAT FLUXES IN CHANNEL # 1

METAL-WATER-REACTION HEAT FLUX (WATT/M²)

FIG. 4.3.10. METAL-WATER-REACTION HEAT FLUXES IN CHANNEL # 8

between FRHS 8-8 and heat slabs in other channels is to reduce the net (outward) radiation heat flux. As a matter of fact, Figure 4.3.8 indicates that total radiation heat flux for FRHS 8-8 may even become negative in the early stage of blowdown.

- (d) Metal-Water-Reaction (MWR) heat flux increases with the increase of cladding temperature and decreases with the increase of cladding oxide penetration depth. As a result, following the incipience of film boiling, MWR heat flux generally responds with an initial surge and then decreases with increasing cladding oxidation. As shown in Figures 4.3.9 and 4.3.10, radiation heat transfer can cause a large reduction of MWR heat flux. This phenomenon reflects the exponential dependence of MWR heat flux on temperature and the fact that radiation heat transfer is very effective in reducing cladding temperature (see next subsection). Generally speaking, MWR heat flux is not an appreciable part of total heat flux if radiation heat transfer is considered.

4.4 Fuel Rod Temperature Distribution

Fuel rod temperature distribution will be specified by the following parameters:

- (1) Fuel centerline temperature,
- (2) Fuel surface temperature,
- (3) Cladding inside surface temperature,
- (4) Cladding outside surface temperature.

The histories of these parameters are given in Figures 4.4.1 through 4.4.8. From these figures, one observes that:

- (a) Run #1 and Run #2 indicate that, for FRHS 1-9, the computed values of fuel centerline temperature, fuel surface temperature, cladding inside surface temperature and cladding outside surface temperature show an appreciable decrease when radiation heat transfer is considered. In the later stage of blowdown, the decrease is about 70 K for fuel centerline temperature and is about 100 K for other temperatures. It is seen that, without considering radiation heat transfer (Run #2), the temperatures at fuel surface, cladding inside and outside surfaces increase persistently even after $t = 10$ sec. If radiation heat transfer is considered (Run #1), computed temperatures at these spots level off at $t = 10$ sec. The cause of these differences is that, for FRHS 1-9, radiation heat flux becomes a major contribution of total heat flux at the later stage of blowdown (see (b) of Subsection 4.3) and helps to hold down fuel rod temperatures at these spots.

For FRHS 8-8, the cladding outside temperature generally is much lower than that of FRHS 1-9 and the effect of radiation heat transfer is also smaller. The peak cladding temperature of the peripheral fuel rod continues to increase after $t = 10$ sec since radiation heat transfer is insignificant.

- (b) The effect of crossflow on cladding surface temperature of the hottest fuel rod heat slab is small when compared with that of radiation heat transfer. As shown in Figure 4.4.7, cladding outside surface temperature of the hottest fuel rod segment reduces no more than 30 K when the crossflow effect is considered. When the effect of radiation heat transfer is also considered, a further reduction of 100 K can be realized.

FIG. 4.4.1. FUEL ROD CENTER LINE TEMPERATURES IN CHANNEL # 1

FIG. 4.4.2. FUEL ROD CENTER LINE TEMPERATURES IN CHANNEL # 8

FIG.4.4.3. FUEL SURFACE TEMPERATURES IN CHANNEL # 1

FIG. 4.4.4. FUEL SURFACE TEMPERATURES IN CHANNEL # 8

CLADDING INSIDE SURFACE TEMPERATURE (K)

FIG. 4.4.5. CLADDING INSIDE SURFACE TEMPERATURES IN CHANNEL # 1

CLADDING INSIDE SURFACE TEMPERATURE (K)

FIG. 4.4.G. CLADDING INSIDE SURFACE TEMPERATURES IN CHANNEL # 8

CLADDING OUTSIDE SURFACE TEMPERATURE (K)

FIG. 4.4.7. CLADDING OUTSIDE SURFACE TEMPERATURES IN CHANNEL # 1

FIG. 4.4.8. CLADDING OUTSIDE SURFACE TEMPERATURES IN CHANNEL # 8

Generally speaking, radiation heat transfer is more effective than crossflow in the reduction of temperature at any spot in a hot fuel rod segment. These conclusions are based on the assumption of uniform core fluid boundary conditions. Future calculations are planned to evaluate the effect of non-uniform core boundary conditions on these conclusions.

Figures 4.4.2, 4.4.4, 4.4.6 and 4.4.8 all indicate that the effect of crossflow on the reduction of temperature at any spot in FRHS 8-8 is negligible.

5.0 CONCLUSIONS AND RECOMMENDATIONS

A preliminary study of the potential influence of crossflow and radiation heat transfer on LOFT fuel rod response for the L2-4 test has been completed. The principal conclusions reached as a result of this effort can be summarized as follows:

- (a) In the later stage of blowdown (about 12 sec after the initiation of blowdown), radiation heat flux of the hottest fuel rod segment becomes comparable with convective heat flux (both are in the range of 50000 watt/m^2). Contrarily, radiation heat flux of a peripheral fuel rod segment is only a minor part of total heat flux.
- (b) The radiation heat flux is responsible for 100 K reduction of peak cladding temperature during blowdown. If the minor effect of crossflow is also considered, the total reduction of peak cladding temperature during blowdown is about 125 K.
- (c) Except at the core outlet, crossflow is negligible compared with axial flow.
- (d) Radiation heat transfer helps to hold down cladding temperature and, in turn, metal-water-reaction heat generation is reduced substantially.
- (e) As discussed in Appendix B, improvements to the MOXY/SCORE code hydraulic numerical scheme have been made. The improvements have significantly reduced computer running time (a reduction of 70% was observed) for LOFT predictions.

The results of this preliminary study indicate that the strong influence of 3-D effects (particularly the effect of radiation heat transfer) on fuel rod response does exist for LOFT L2-4 test. For future efforts on this subject, it is recommended that:

- (a) The effects of nonuniform flow distribution at inlet (or outlet) be evaluated,
- (b) A more realistic gap conductance model be used,
- (c) A more detailed radiation calculation be performed for central module. The fuel rods and guide tubes should be represented discretely,
- (d) Effort be made to linearize pressure iteration in MOXY/SCORE hydraulic calculation. The success of this effort will result in a large reduction of MOXY/SCORE running time.
- (e) Extend the MOXY/SCORE calculation beyond the blowdown phase if the effort described in (d) makes it feasible.

APPENDIX A.

VIEW FACTOR CALCULATION

The view factors (or configuration factors) between heat slabs at the same axial level are among the input data for MOXY/SCORE radiation heat transfer calculation. The calculation of these factors will be discussed in this appendix.

For MOXY/SCORE calculation, LOFT reactor core is modeled by 16 flow channels and 15 axial levels. Each fluid cell in Channels #1 through #15 (see Figs. A.1 & A.2) is the heat sink of at least one heat slab (which represents a collection of 21.81 identical fuel rod segments at the same axial level). In addition to this fuel rod heat slab, each fluid cell in channel #4, #8, #12, #13, #14, & #15 is also a heat sink for a canister heat slab. Because (1) the view factors between heat slabs at one axial level are identical to those at another axial level, and (2) the radiation heat transfer calculation is limited to heat slabs at the same axial level, one only needs to compute view factors between heat slabs at the first axial level (from now on, it will be assumed implicitly that heat slabs mentioned are at the first axial level).

Because a fuel rod heat slab is a collection of identical fuel rod segments (a fuel rod segment is the portion of a fuel rod which belongs to an axial level), the calculation of view factors between fuel rod heat slabs must begin with the calculation of view factors between individual fuel rod segments. Fig. A.3 depicts a typical fuel rod segment A and some of its surrounding fuel rod segments. The method of fuel rod segments view factor calculation can be briefly described by:

1	5	9	13
2	6	10	14
3	7	11	15
4	8	12	16

FIG. A.1. FLOW CHANNELS
FOR MOXY/SCORE MODEL

1			
2	5		
3	6	8	
4	7	9	

↑ 10 ↑ 11 ↑ 12

FIG. A.2. HEAT SLABS
AT FIRST AXIAL LEVEL.
(ONLY INDEPENDENT HEAT
SLABS ARE NUMBERED. HEAT
SLABS #1 THROUGH #9 ARE
FUEL ROD HEAT SLABS. HEAT
SLABS #10, #11 & #12 ARE
CANISTER HEAT SLABS.)

M	N	O	P	Q
L	C	D	E	R
K	B	A	F	S
J	I	H	G	T
Y	X	W	V	U

FIG. A.3. A TYPICAL FUEL ROD SEGMENT A
AND ITS SURROUNDING SEGMENTS

FIG. A.4 GEOMETRIC SPECIFICATION OF
FUEL ROD SEGMENTS

- (a) The view factors from segment A to segments on its first adjacent layer will be calculated analytically by assuming that (1) the diameters of all fuel rod segments are always identical, and time independent, and (2) the ratio between fuel rod segment height and its diameter is so large (the real value is 10.43) that they can be considered as infinite long cylinders. If one defines that:

$F_{A \rightarrow E}$ = view factor from segment A to segment E

$F_{A \rightarrow F}$ = view factor from segment A to segment F and so on,

then it is seen that:

$$F_{A \rightarrow F} = F_{A \rightarrow H} = F_{A \rightarrow B} = F_{A \rightarrow D} \equiv x \quad (A.1) \text{ and}$$

$$F_{A \rightarrow E} = F_{A \rightarrow G} = F_{A \rightarrow I} = F_{A \rightarrow C} \equiv y \quad (A.2)$$

- (b) Because the diameters of all fuel rod segments are always identical, it can be seen that:

$$F_{A \rightarrow K} = F_{A \rightarrow M} = F_{A \rightarrow O} = F_{A \rightarrow Q} = F_{A \rightarrow S} = F_{A \rightarrow U} = F_{A \rightarrow W} =$$

$$F_{A \rightarrow Y} = 0 \quad (A.3)$$

- (c) It is assumed that the view factor between segment A and any segment which is located beyond the second adjacent layer of segment A is zero. This assumption coupled with Eqs. A.1, (A.2), & (A.3) and the fact that $F_{A \rightarrow A} = 0$ implies that:

$$F_{A \rightarrow L} = F_{A \rightarrow N} = F_{A \rightarrow P} = F_{A \rightarrow R} = F_{A \rightarrow T} = F_{A \rightarrow V} = F_{A \rightarrow X} =$$

$$F_{A \rightarrow J} = \frac{1}{8} (1-4x-4y) \quad (A.4)$$

Fig. A.4 gives the dimensions needed to calculate x and y defined in Eqs. (A.1) - & (A.2). Using the crossed string method^[4], x and y are found to be

$$x = 0.1262 \text{ and } y = 0.0864 \quad (\text{A.5})$$

Eqs. (A.1) - (A.5) and the assumption that any segment can only see segments on its first and second adjacent layers enable one to calculate all the views factors between fuel rod segments.

The next step is to calculate the view factors between heat slabs. Fig. A.5 depicts three neighboring fuel rod heat slabs A, B, & C. Each heat slab is a 5 x 5 square matrix of fuel rod segments. If one defines

$$F_{A-5 \rightarrow B-1} = \text{view factor from segment A-5 to segment B-1}$$

$$F_{A \rightarrow B} = \text{view factor from heat slab A to heat slab B}$$

then, by the definition of view factor, one has

$$F_{A \rightarrow B} = \frac{1}{n^2} \sum_{i,j=1}^{n^2} F_{A-i \rightarrow B-j} \quad (\text{A.6})$$

$$F_{A \rightarrow C} = \frac{1}{n^2} \sum_{i,j=1}^{n^2} F_{A-i \rightarrow C-j} \quad (\text{A.7})$$

Here it is assumed that each heat slab is a $n \times n$ square matrix of fuel rod segments. Eqs. (A.1) - (A.7) coupled with the assumption that any fuel rod segment can not see segments which are beyond the second adjacent layer imply that:

FIG. A.5. NEIGHBORING FUEL ROD HEAT SLABS

$$F_{A \rightarrow B} = \frac{1}{n^2} \left[\frac{(3n-4)}{4} - (n-2)(2x+y) \right] \quad n = 4, 5, 6, \dots \quad (A.8)$$

$$F_{A \rightarrow C} = \frac{1}{n^2} [0.5 - 2x - y] \quad n=2, 3, 4, \dots \quad (A.9)$$

Each fuel rod heat slab shown in Fig. A.2 is a collection of 21.81 fuel rod segments. To calculate the view factors between these heat slabs, it will be assumed that Eqs. (A.8) & (A.9) can be used for any positive number $n \geq 4$. Let $n^2 = 21.81$ ($n = 4.67$), $x = 0.1262$ and $y = 0.0864$, then Eqs. (A.8) and (A.9) imply that (see Fig. A.2):

$$F_{2 \rightarrow 5} = \frac{1}{21.81} \left[\frac{(3 \times 4.67 - 4)}{4} - (4.67 - 2)(2 \times 0.1262 + 0.0864) \right] = 0.0732 \quad (A.10)$$

$$F_{2 \rightarrow 6} = \frac{1}{21.81} (0.5 - 2 \times 0.1262 - 0.0864) = 0.00739 \quad (A.11)$$

By symmetry, it is seen that

$$F_{2 \rightarrow 2} = 1 - 4 [F_{2 \rightarrow 5} + F_{2 \rightarrow 6}] = 0.678 \quad (A.12)$$

Other view factors between fuel rod heat slabs shown in Fig. A.2 can be inferred by symmetric considerations.

The view factors between fuel rod heat slabs and canister heat slabs can be estimated as follows:

$$F_{4 \rightarrow 10} = \frac{1}{4} [1 - F_{4 \rightarrow 4}] = \frac{1}{4} [1 - 0.678] = 0.0805 \quad (A.13)$$

$$F_{4 \rightarrow 11} = \frac{1}{4} [1 - F_{4 \rightarrow 4} - 2 \times F_{4 \rightarrow 10} - 2 \times F_{4 \rightarrow 7}] = \frac{1}{2} F_{2 \rightarrow 6} = 0.0037 \quad (A.14)$$

$$A_4 F_{4 \rightarrow 10} = A_{10} F_{10 \rightarrow 4} \quad [A_i = \text{surface area of heat slab \# } i]$$

$$F_{10 \rightarrow 4} = (A_4/A_{10}) F_{4 \rightarrow 10} = \frac{\pi \times 0.422 \times 21.81}{2.79} \times 0.0805 = 0.834 \quad (\text{A.15})$$

$$A_4 F_{4 \rightarrow 11} = A_{11} F_{11 \rightarrow 4}$$

$$F_{11 \rightarrow 4} = (A_4/A_{11}) F_{4 \rightarrow 11} = \frac{\pi \times 0.422 \times 21.81}{2.79} \times 0.0037 = 0.0383 \quad (\text{A.16})$$

Other view factors between fuel rod heat slabs and canister heat slabs can also be inferred by symmetric consideration.

The view factors calculated in this appendix are summarized in Table A.1. Table A.2 lists the original view factor input data. The input data were calculated with less rigor and were slightly different from those given in Table A.1. Because the error due to these slight differences would be small, and because the expense in recomputing would be large, little value was seen in recomputing the problem.

i	j	$F_{i \rightarrow j}$	i	j	$F_{i \rightarrow j}$
1	1	0.678	1	5	0.0074
2	2	0.678	2	6	0.0074
3	3	0.678	3	5	0.0074
4	4	0.678	3	7	0.0074
5	5	0.678	4	6	0.0074
6	6	0.678	5	1	0.0074
7	7	0.678	5	3	0.0074
8	8	0.678	5	8	0.0074
9	9	0.678	6	4	0.0074
1	2	0.0732	6	2	0.0074
2	1	0.0732	6	9	0.0074
2	5	0.0732	7	3	0.0074
2	3	0.0732	7	8	0.0074
3	2	0.0732	8	7	0.0074
3	6	0.0732	8	5	0.0074
3	4	0.0732	9	6	0.0074
4	3	0.0732	4	10	0.0805
4	7	0.0732	7	11	0.0805
5	2	0.0732	9	12	0.0805
5	6	0.0732	4	11	0.0037
6	3	0.0732	7	10	0.0037
6	5	0.0732	7	12	0.0037
6	8	0.0732	9	11	0.0037
6	7	0.0732	10	4	0.834
7	4	0.0732	11	7	0.834
7	6	0.0732	12	9	0.834
7	9	0.0732	10	7	0.038
8	6	0.0732	11	4	0.038
8	9	0.0732	11	9	0.038
9	7	0.0732	12	7	0.038
9	8	0.0732			

TABLE A.1. VIEW FACTORS GIVEN BY EQS (A.10)-(A.16)

VIEW FACTOR INPUT DATA

I	J	FIJ
1	1	.6834
2	2	.6834
3	3	.6834
4	4	.6834
5	5	.6834
6	6	.6834
7	7	.6834
8	8	.6834
9	9	.6834
1	2	.0714
2	1	.0714
2	5	.0714
2	3	.0714
3	2	.0714
3	6	.0714
3	4	.0714
4	3	.0714
4	7	.0714
5	2	.0714
5	6	.0714
6	3	.0714
6	5	.0714
6	8	.0714
6	7	.0714
7	4	.0714
7	6	.0714
7	9	.0714
8	6	.0714
8	9	.0714
9	7	.0714
9	8	.0714
1	5	.0077
2	6	.0077
3	5	.0077
3	7	.0077
4	6	.0077
5	1	.0077
5	3	.0077
5	8	.0077
6	4	.0077
6	2	.0077
6	9	.0077
7	3	.0077
7	8	.0077
8	7	.0077
8	5	.0077

VIEW FACTOR INPUT DATA (cond)

I	J	FIJ
9	6	.0077
4	10	.0791
7	11	.0791
9	12	.0791
4	11	.0038
7	10	.0038
7	12	.0038
9	11	.0038
10	4	.9500
11	7	.9500
12	9	.9500
10	7	.0250
11	4	.0250
11	9	.0250
12	7	.0250

TABLE A.2. VIEW FACTOR INPUT DATA

APPENDIX B

NUMERICAL CONSIDERATION

This appendix is devoted to discussion of the numerical problems of the MOXY/SCORE hydraulic calculation. The hydraulic calculation has been the major cause of MOXY/SCORE slow running time. To cut short MOXY/SCORE running time, it is imperative to improve the numerical scheme of the hydraulic calculation.

Basically, the numerical technique used in hydraulic calculation is based on the MAC method for incompressible flow as modified by Hirt and Cook^[5] and the ICE technique for compressible flow^[6]. The essential features of these two methods are: (1) the continuity equation is implicit in density and velocity, (2) the linear momentum equation is implicit in pressure, and (3) the energy is purely explicit. The procedure of the hydraulic calculation can be summarized by:

- (1) The new specific internal energy is calculated explicitly with energy.
- (2) The tentative new velocities are calculated with the momentum equation using the latest tentative new pressure.
- (3) The tentative new density is calculated with the equation of state using the new internal energy and the latest tentative new pressure.
- (4) With the help of the continuity equation and the equation of state, the tentative new pressure is calculated in terms of the latest tentative new pressure and density.
- (5) Steps 2 through 5 are repeated until some convergence criterion is satisfied. The latest tentative new fluid quantities are then set to be new fluid quantities.

Because the continuity equation is implicit in density and velocity, the hydraulic calculation is very stable, i.e., the calculation will converge even if a large time-step is used. However, due to the same reason, the pressure iterations described in Step (4) become nonlinear and, even for a time-step well under the limit set by Courant's stability condition, it generally takes several hundred iterations before the convergence criterion is satisfied. In other words, the use of a larger time-step is allowed but not practical for a nonlinear pressure iteration scheme.

An improved version of the nonlinear pressure iteration scheme was used in this study. The essence of this improvement is to devise a method for making an accurate first guess for pressure and enable a faster convergence of pressure iteration. Although this improvement has cut MOXY/SCORE running time by as much as 70%, it is not a final solution because the reduced running time is still very substantial and its success is dependent of the smoothness of pressure transient.

A better way to go around the difficulties encountered by a nonlinear pressure iteration is to make the velocity the only implicit quantity in the continuity equation. This modification will linearize the pressure iteration scheme and result in a much faster convergence.

APPENDIX C

RELATIONSHIP BETWEEN CONVECTIVE AND RADIATIVE
HEAT TRANSFER TO FLUID

In MOXY/SCORE, convective heat transfer is calculated using the same logic and, with one exception noted later, the same correlations as are used in RELAP4/MOD5^[3]. Radiative heat transfer to the fluid is considered to be additive to convective heat transfer, in terms of both removing energy from the fuel rod and adding energy to the fluid. Since no method exists for partitioning radiative and convective heat transfer to water in experiments, it is clear that correlations for convective heat transfer based on data from such experiments must include radiative heat transfer effects, as well. Therefore, the possibility exists that radiative heat transfer to fluid is duplicated in MOXY/SCORE.

The model for radiation to fluid is discussed elsewhere^[7]. This model is based on the assumption that liquid water completely absorbs thermal radiation. Thus, for void fractions less than 0.1, radiation to fluid is neglected. For void fractions equal to or greater than 0.1, the transmission fraction for radiation passing between surfaces is obtained by multiplying the value computed for a void fraction of 1.0 by the void fraction, thus assuming a dispersed flow pattern. Also, the radiation term of the free convection and radiation regime is deleted whenever radiation to fluid is considered, avoiding an obvious duplication.

In the pre-CHF heat transfer regimes - forced convection to subcooled liquid, nucleate boiling, and forced convection vaporization - the fuel rods are wetted, and the surface temperatures are close to that of the fluid. Under these conditions, radiation to fluid is small due to the small temperature difference between fuel rod and fluid. Therefore, whatever duplication exists occurs at a negligible level.

In the post-CHF and dryout regimes - transition boiling, stable film boiling, low-flow, low-void-fraction film boiling, low-flow, high-

void-fraction free convection and radiation, and forced convection to single-phase steam - more potential for duplication exists. In transition boiling and low-flow, low-void-fraction film boiling, the difference between rod surface temperatures and the fluid temperature is quite low, and void fractions tend to be low, minimizing the computed values of radiation to steam. Only in high-void-fraction conditions, with high differences between fuel rod temperature and fluid temperature, is radiation to steam of high magnitude. As was mentioned earlier, the radiation portion of the free convection and radiation regime is deleted. In high-void-fraction stable film boiling and single-phase forced convection, the possibility of duplication exists. Further study is necessary to evaluate the significance of this duplication and if a problem exists, to identify a solution. To provide some perspective, heat transfer correlations have significant (~25 percent) uncertainty, so effect of duplication is likely to be of the order of the uncertainty. Therefore, it is reasonable to conclude that the significance of duplication is likely to be minimal.

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