A Three-Dimensional Code for Thermonuclear Package Design to Transport Spent-Fuel Elements

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ABSTRACT

A three-dimensional code named thermonuclear design for spent-fuel package (TNDSFP) was developed based on the thermonuclear transient analysis of spent-fuel elements located in a squared arrayed dry package. The code comprises the analysis of both nuclear and thermal models to get the time-dependent heat generation source strength and the transient thermal analysis of the spent-fuel elements during transportation. Both models were matched to meet the safety requirement to yield the design parameters for a square lattice dry transportation package. The code has been tested against parametric study for sake of debugging the program package. In this investigation, the code models description, characteristics, and overall organization with supporting subroutines are presented.

INTRODUCTION

The proper evaluation of the consequences of transportation of spent-fuel is of great importance in nuclear safety. The long period of transportation may cause heating up of the interior and/or fuel surface temperatures possibly leading to an accident. Spent-fuel elements, when removed from the reactor, are highly radioactive. Besides it contains some residual heat sources. So they are kept in storage for some time under water to remove some of this residual heat . After a reasonable cooling time, the spent-fuel elements could be loaded in heavily shielded packages to reprocessing plants. The heat generated by radioactive decay must be dissipated, otherwise serious difficulties-such as rupturing or melting of the spent elements with a consequent dispersal of radioactivity or a possible formation of a critical configuration-will result from the extensive temperatures within the package structure or through its contents. Thus the heat transfer in the spent-fuel package has to be considered as a significant safety problem.

THERMONUCLEAR MODEL AND ITS NUMERICAL SOLUTIONS

The nuclear analysis incorporates evaluation of the local instantaneous volumetric heat generation source strength in the spent- fuel element arising from the different heat sources existing after fuel removal from the reactor. These sources are residual fission heat , fission products, decay heat, and decay heat of actinides. The heat source due to residual fission heat is obtained through the solution of the space-dependent reactor kinetics equations. The power generation continues after the fission source stops due to the fission products decay. This heat source varies with a rate depending on the reactor core operating history. The decay heat source is fitted to a polynomial of 11 exponentials. The energy release data for infinite operating time have been used in the data fitting. In the present work, the energy release from fission products decay is considered by including the important radioactive actinides(U-239 and Np-239) that are produced by radioactive neutron capture in U-238. The decay heat contributions from these isotopes are obtained from the ANS standard (ANS, 1977) for infinite operating times. The differential equations of the three heat sources are solved numerically using Adam's methods. The algorithms of all Adam's methods are obtained by using the PECE method of the predictor-corrector (Shampine & Gordon, 1975).

The heat transfer processes in the dry package include radiant heat transfer besides the heat transfer by thermal conduction.The radiant heat transfer in the spent-fuel dry cask is considered the dominant heat dissipation process for the resulting temperature distribution. These calculations were made using the net radiation method (McAdams, 1985).The fuel heat transfer model calculates the internal temperature distribution within the fuel solid material considering the effects of both the axial conduction and the temperature-dependence of fuel thermal conductivity. The heat flow equation is derived by considering the variation of the heat quantity in an arbitary volume during a time interval. For the heat flow problems of the present code, the equations are solved in three-dimensional cylindrical geometry, which is reduced to linear partial differential equation using Kerchief's Transformation (Ozisik, 1980) and then solved numerically using a combination of orthogonal collocation technique (Finleyson, 1972) and finite difference technique. For the calculation of the initial stored energy and the transient heat transfer across the gap, the thermal conduction of the gap is considered as a function of the burnup taking into account the composition and pressure of the gases within the spent-fuel element and the initial cold gap dimension with tolerances. So the dynamic gap model used in this code considers the gap heat transfer coefficient as a function of two compounds including the gap gas conductance and the fuel-clad thermal radiation, and assuming a conductance through points of contact of the cladding and fuel. The gas conductivity of the gap gas mixture is computed as a function of gap temperature and based on the rigorous theory (Muckenfuss & Curtiss, 1958). For the heat conduction through the clad, the transient energy balance for lumped clad in finite difference form is used and then transformed to get the implicit time derivative and the explicit axial conduction using Karchief's transformation (Ozisik, 1980). Implicit temperatures are then evaluated using the truncated Taylor Series (Lienhard,1981). The three equations governing the heat conduction through fuel interior to clad surface are combined together to yield a matrix equation for the transformed temperature, which is solved by iterative Gauss-Seidel procedure (Ozisik, 1980). The iteration is terminated when convergence is satisfied.

CODE DESCRIPTION AND OVERALL ORGANIZATION

The TNDSFP code consists of some computer modules developed to evaluate the thermonuclear analysis of the spent-fuel element in order to obtain the effect of transportation time on both the spent-fuel interior and clad surface temperatures at different pitches after the spent fuel is kept cooled for different periods of time, within the International Atomic Energy Agency (IAEA) frame of regulations for the safe transport of radioactive materials. It is developed with the intent of producing a simple, relatively accurate and efficient prediction scheme. It is characterized by modularity and compatibility with PC computers. Thus, some specialized coding has been segregated to subprograms that perform a specific function. Table 1 gives the TNDSFP code supporting subroutines' functions. Figure 1 illustrates the block flow chart of the code.

Table 1. TNSSFP Supporting Subroutines

NAME	FUNCTION	NAME	FUNCTION
FIC	Fuel Initial Conditions	NA	Nuclear Analysis Program
RFH	Residual Fission Heat	FPDH	Fission Product Decay Heat
DHOA	Decay Heat of Actinides	NAM	Nuclear Analysis Module
TNMM	Thermonuclear Matching Module	THA	Thermal Analysis Program
TR	Thermal Radiation Module	HC	Heat Conduction Program
HCG	Heat Conduction Through Gap	COP	Code Output
HCC	Heat Conduction in Clad	THAM	Thermal Analysis Module
HCFI	Heat Conduction Through Fuel Interior		

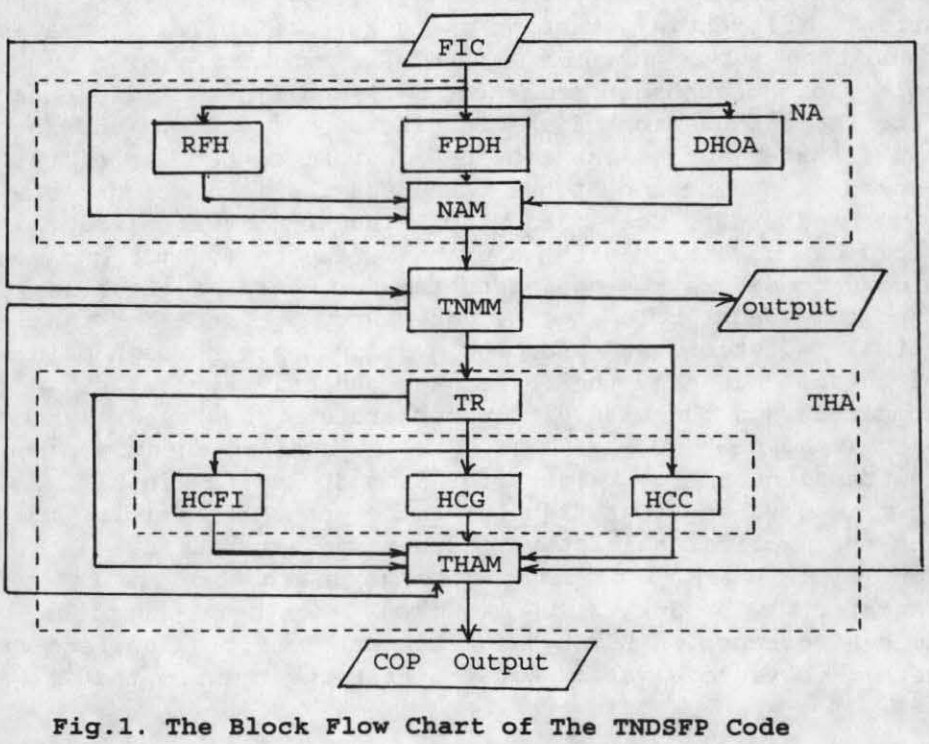

Fig.1. The Block Flow Chart of The TNDSFP Code

As shown in the figure, the TNDSFP code consists of :
1. A nuclear analysis for the variation of the transient power profile, which gives the time-dependent volumetric heat generation source strength. This is performed through the NA program which contains the RFH, FPDH, DHOA, and NAM modules.
2. A transient thermal analysis of the spent-fuel element during transportation time to get the details transient temperature distribution (through fuel interior and on clad surface) for any spent-fuel element located at any place in the square arrayed package matrix. This is performed through the THA program, which contains TR, HCFI, HCG, HCC, and THAM modules. These two models are matched to meet the safety requirement to yield the design parameters for the square lattice dry transportation package. It is performed through the TNMM module. The processing of all TNDSFP input information is accomplished by direct input data statement retrieved from the Fuel Initial Condition FIC subroutine and data information obtained as inherent outputs from other modules through TNMM.

CONCLUSIONS

1. This investigation has resulted in the description of the developed TNDSFP code, summing up its charaterization and overall organization and outlining its block flow chart.
2. The code is used to obtain the thermonuclear transient analysis of any type of spent-fuel element (with any specified percentage of burnup) stowed in a square lattice dry package.
3. It could review the safety criteria of any already-existing transport square matrix dry package of a specified design.
4. The code is flexible and easy to be manipulated. It is modular by function, compatible with PC computers, and has an efficient prediction scheme.

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