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Real Time Simulation of the Containment

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Abstract

Since the accidents at Three Mile Island and Chernobyl, more stringent requirements have been imposed on the containment system for full replica nuclear power plant simulators. The software model must be able to accurately respond under all normal operations and especially under abnormal conditions, including leaks and breaks of any size in any location in the Reactor Building.

A multiple node containment model, based on first engineering principles, was developed at CAE; it has already been successfully implemented on a number of simulators. An important feature of the model is its versatility. It is generic in that the techniques and methods used in the modelling can be easily implemented for any type of containment. It is also plant specific in that the nodalization is dictated by the physical structure, geometry, and dimensions of the reference plant. Thus, nodalization changes due to addition or removal of equipment or leaks can be done in a straightforward manner.

The techniques used to develop this model with all its features are presented in this paper. Typical transient results obtained for various LOCA's are included.

Introduction

The primary purpose of the containment is to minimize the uncontrollable release of radioactive materials to the environment. The Three Mile Island and Chernobyl incidents have increased awareness for the requirements for more sophisticated nuclear power plant simulators, capable of reliably reproducing containment response under all conditions.

Following consultations with the customers, a containment model was developed at CAE to provide the operators with maximum training values and to ensure that the criteria, discussed below, are achieved.

This model is versatile. It was originally developed for the containment of the St-Lucie simulator and has since been updated and implemented on the Davis-Besse, Crystal River, and San Onofre simulators. It is designed in a way that could easily be modified to fit the specific requirements of any containment for any type of power plant as dictated by the configuration, dimensions, and structures. The number of control volumes and configuration required to correctly duplicate the actual containment response and the proper temperature and radiation stratifications is easily incorporated. Moreover, this model was developed with the end user in mind, it is structured in a systematic manner, thus making it easy to maintain and modify.

Most importantly, the model is reliable in reproducing containment response under all normal and abnormal conditions. Reliability is achieved through the application of first principles. This means that mass balances are performed for each component in each vapour space control volume; energy balances are performed in each vapour space control volume for condensibles and for noncondensibles; and conservation of momentum is used to determine the flow rates between vapour space control volumes. The model is able to faithfully reproduce the various phenomena which might occur in the containment. These phenomena include flashing rates of all possible leaks, evaporation from the liquid spaces, and condensation in the coolers, on the walls, due to rain out (inability of the air to support any more water vapour), and due to the operation of the containment sprays.

Model Nodalization

The nodalization is plant specific in that the number of control volumes and their locations depend on the physical structure and restrictions, the equipment and their locations, and the anticipated flow paths. A typical PWR containment is shown in figure 1; this containment is divided into 5 control volumes (CV's) as presented in figure 2: CV-1 covers the area around the reactor vessel inside the primary shield, CV-2 covers the dome area inside the containment, CV-3 and CV-4 present the steam generators compartments, and finally CV-5 covers the annular space between the secondary shield and the containment walls. Figure 3 shows the corresponding nodal diagram and the possible flow paths within the containment.

Figure 1: Typical PWR Containment

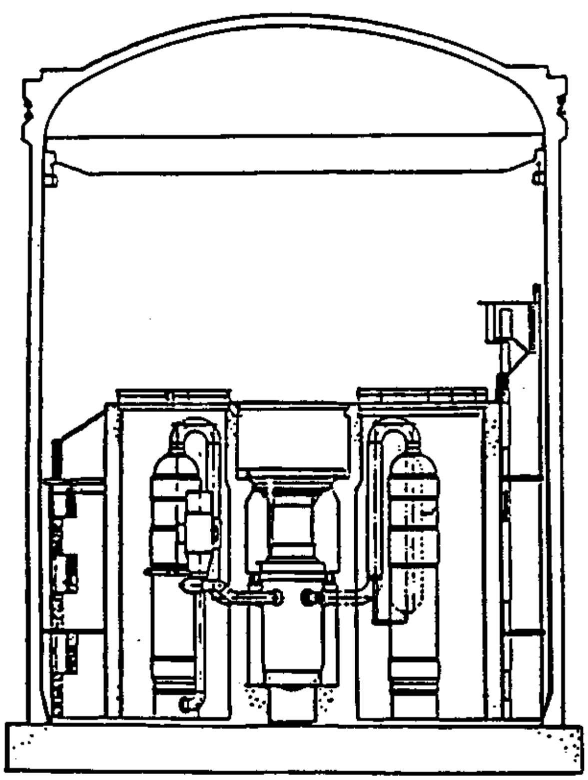


Figure 2: Containment Control Volumes

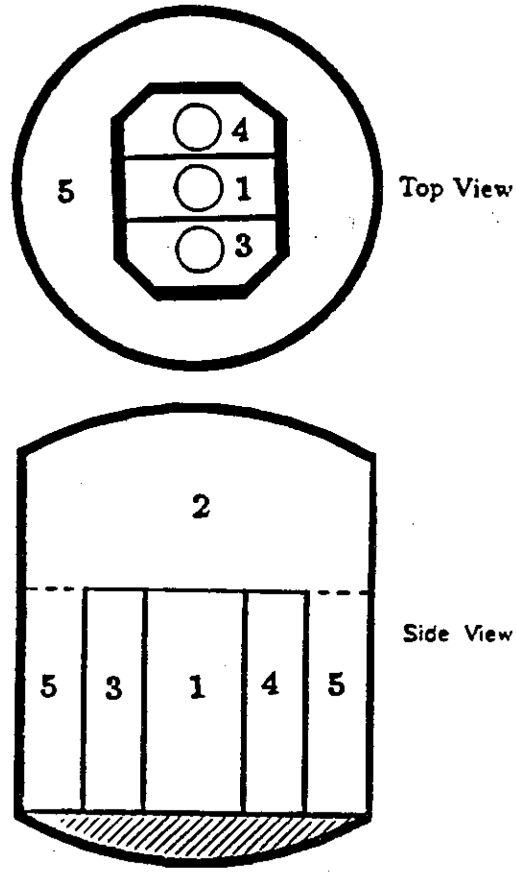
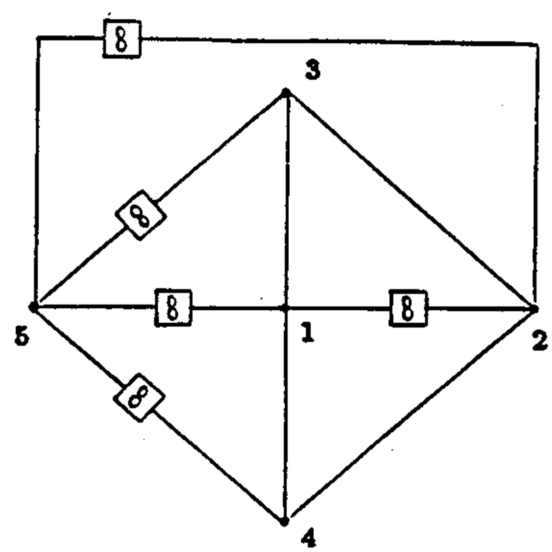


Figure 3: Nodal Diagram



Conservation of Component Mass

A mass balance is performed for each component (nitrogen, oxygen, hydrogen, and water vapour) in each vapour space CV taking into account all possible flows, including flows between the CV's, and leaks and breaks into the containment.

Consider a vapour space CV "k" with neighbouring CV's "j" and "l". The component mass "i" in CV "k" can be described by the mass balance equation:

$$\frac{dM_{ki}}{dt} = \sum W_{jk}X_{ji} + \sum W^e_{jk}X^e_{ji} - \sum W_{kl}X_{ki} - \sum W^e_{kl}X_{ki} \quad (1)$$

where,

M_{ki}	Mass of component i in CV k
W_{jk}	Internal mass flow rate from CV j to k
X_{ji}	Mass fraction of component i in internal CV j
W^e_{jk}	Mass flow rate from external CV j to internal CV k
X^e_{ji}	Mass fraction of component i in external CV j
W_{kl}	Internal mass flow rate from CV k to l
X_{ki}	Mass fraction of component i in internal CV k
W^e_{kl}	Mass flow rate from internal CV k to external CV l

By defining,

$$X_{ji} = \frac{M_{ji}}{M_j} \quad (2)$$

where,

M_{ji}	Mass of component i in CV j
M_j	Total mass in CV j

then substituting the expression for X_{ji} in equation (1), and using backwards differences, the conservation of mass equation is given in matrix form by:

$$AM_{ki} = S_{kl} \quad (3)$$

where,

A	Mass matrix
S_{kl}	Source vector

Conservation of Energy

The conservation of energy equation is applied separately for the condensibles (water vapour) and for the noncondensibles (nitrogen, oxygen, and hydrogen combined) for each CV and is consistent with the component mass balances; the noncondensible gases are assumed to be in thermal equilibrium. Since the time constant of the enthalpy response is always greater than the module iteration rate, the equations may be solved sequentially without affecting the transient response. The energy balance takes into account all the relevant inflows and their enthalpies, as well as the heat flux and heat exchange with energy sources and heat sinks. Taking backwards differences, the energy balance equation for component "r" in CV "k", where "r" represents either condensibles or noncondensibles, can then be implicitly integrated as:

$$h_{kr} = \frac{h'_{kr}M_{kr} + dt(\sum W_{jk}X_{jr}h_{jr} + \sum W^e_{jk}X^e_{jr}h^e_{jr} + Q_{kr})}{M_{kr} + dt(\sum W_{jk}X_{jr} + \sum W^e_{jk}X^e_{jr})} \quad (4)$$

where,

M_{kr}	Mass of component r in CV k
h_{kr}	Specific enthalpy of component r in internal CV k
h'_{kr}	Value of h_{kr} from previous iteration
X_{jr}	Mass fraction of component r in internal CV j
h_{jr}	Specific enthalpy of component r in internal CV j
X^e_{jr}	Mass fraction of component r in external CV j
h^e_{jr}	Specific enthalpy of component r in external CV j
Q_{kr}	Heat transferred to component r in CV k from external heat sources and sinks. e.g. reactor vessel, walls, coolers, heaters, etc.

Temperature and Pressure Calculation

Having determined the masses and energies in the various CV's, the state of each CV can be found (assuming perfect mixing in each CV).

The enthalpy of the gas/water-vapour mixture is calculated based on the Gibbs-Dalton law (Wark 1983) which states that the mixture enthalpy is simply the sum of the enthalpies of the individual components.

By using the definition of heat capacity at constant pressure which relates temperature to enthalpy and using the partial pressure of vapour to evaluate the saturation conditions, the temperature of each CV is calculated.

$$T_k = \frac{h_{mix} - [h_g(P_{kv}) - Cp_{kv} * T_{sat}(P_{kv})] * X_{kv} - h_{ref} * \sum X_{knci}}{(X_{kv} * Cp_{kv}) + Cp_{knc} * \sum X_{knci}} \quad (5)$$

where,

T_k	Mixture temperature in CV k
h_{mix}	Mixture enthalpy in CV k
$h_g(P_{kv})$	Saturation enthalpy at the partial pressure of vapour in CV k
$T_{sat}(P_{kv})$	Saturation temperature at the partial pressure of vapour in CV k
X_{kv}	Mass fraction of water vapour in CV k
X_{knci}	Mass fraction of noncondensible component i in CV k
h_{ref}	Reference enthalpy evaluated at reference temperature $T_{ref} = 0$
Cp_{kv}	Specific heat of vapour in CV k
Cp_{knc}	Specific heat of noncondensibles (assumed to be nitrogen) in CV k

Having determined the temperature, the partial pressures of the various components in each CV are computed. For the noncondensibles, the partial pressures are calculated from the perfect gas law (Wark 1983); whereas, the partial pressure of the vapour is calculated as a function of density and temperature. The total pressure in each CV is then calculated simply as the sum of the partial pressures of the individual components. Therefore,

$$P_k = \frac{R(T_k + C)}{V_k} \sum \frac{M_{knci}}{M_{wnci}} + P_{kv}(M_{kv}, V_k, T_k) \quad (6)$$

where,

P_k	Total pressure in CV k
P_{kv}	Partial pressure of vapour in CV k
T_k	Mixture temperature in CV k
V_k	Total volume of CV k
M_{kv}	Mass of vapour in CV k
M_{knci}	Mass of noncondensible component i in CV k
M_{wnci}	Molecular weight of non condensible component i
R	Universal gas constant
C	DegF to DegR (DegC to DegK) Conversion Factor

Conservation of Momentum

The mass flow rate of the mixture between the internal CV's is obtained from the conservation of momentum equation. Natural recirculation within the containment is induced by the variation in density between the CV's, while forced recirculation is calculated based on the difference in pressures between the CV's and the no-load pressures of the fans. Combining the two flows yields,

$$W_{jk} = A_{jk}(P_j + Pnl_{jk} - P_k + (\rho_j z_j - \rho_k z_k) * g) \quad (7)$$

where,

A_{jk}	Admittance of link j-k
P_j	Pressure in CV j
Pnl_{jk}	No-load pressure of the fans on link j-k
P_k	Pressure in CV k
ρ_j	Density in CV j
ρ_k	Density in CV k
z_j	Elevation of CV j
z_k	Elevation of CV k
g	Acceleration due to gravity

Flashing, Evaporation and Condensation

The flashing of all leaks inside the primary containment is determined by comparing the enthalpy of the leak flow to the saturation conditions present in the containment. The saturation conditions are evaluated at the partial pressure of the vapour.

$$W_f = \frac{h_l - h_f}{h_{fg}} * W_l \quad (8)$$

where,

W_f	Flashing flow
h_l	Specific enthalpy of leak flow
h_f	Saturated liquid specific enthalpy of containment vapour
h_{fg}	Latent heat of containment vapour
W_l	Leak flow rate

Evaporation from the liquid CV's (sumps, containment floor) is evaluated based on the difference between the saturation pressure of the liquid evaluated at the temperature of the liquid CV and the partial pressure of the vapour in the vapour space above the liquid.

All heat transfer between two mediums is calculated based on the Fourier equation (Holman 1981):

$$Q = U(T_a - T_b) \quad (9)$$

where,

- Q Heat transfer
- U Overall heat transfer coefficient
- T_a Temperature of medium a
- T_b Temperature of medium b

The condensation rate is determined by calculating the heat transfer from the condensible fluid at the saturation temperature to the cooling medium. The condensation rate is then given by:

$$W_c = \frac{Q_l}{h_{fg}} \qquad (10)$$

where,

- W_c Condensation flow rate
- Q_l Latent heat transfer

Another type of condensation, which is called rain out, is considered when the containment space becomes saturated and can no longer support the amount of vapour present in the atmosphere. The condensation rate is given by

$$W_r = M_{kv} * (1 - X_k) * Cr \qquad (11)$$

where,

- W_r Condensation flow rate due to rain out
- X_k Quality of vapour in CV k
- Cr Rain out constant

Radiation

Apart from the mass components in the CV's that contribute to the total pressure calculations, radioactive species are transported in the vapour space as mass fractions. Different species are transported by different mediums. Since noble gases are not expected to condense into the liquid CV's to any extent, they are transported in the noncondensibles. Other species such as iodine, however, are expected to follow the condensing water vapour into the liquid spaces and also to evaporate from the liquid spaces. These species are transported in the condensibles. The activity equation can be written in a similar manner to the enthalpy equation as follows:

$$A_{kq} = \frac{A'_{kq} M_{kq} + dt(\sum W_{jk} A_{jq} + \sum W^e_{jk} A^e_{jq})}{M_{kq} + dt(\sum W_{jk} + \sum W^e_{jk} + L_q M_{kq})} \qquad (12)$$

where,

- M_{kq} Mass of species q in CV k
- A_{kq} Specific activity of species q in internal CV k
- A'_{kq} Value of A_{kq} from previous iteration
- A_{jq} Specific activity of species q in internal CV j
- internal CV to k
- A^e_{jq} Specific activity of species q in external CV j
- L_q Decay constant of species q

Liquid Control Volumes

The hydraulic masses in the liquid CV are determined using conservation of mass:

$$M_l = M'_l + dt(\sum W_i - \sum W_o) \qquad (13)$$

where,

- M_l Mass of liquid
- M'_l Value of M_l from previous iteration
- W_i Flow into liquid CV
- W_o Flow out of liquid CV

Note that the evaporation and condensation to and from the vapour space CV's are accounted for in the outflow and inflow terms of the above equation.

Enthalpy of the liquid CV's is calculated from an energy balance.

Test results

Some representative test results will now be presented to indicate the effective implementation of the model on the Crystal River Unit 3 Training Simulator. Test results for a large LOCA (full line break) will be displayed and discussed. These results will then be compared to the expected results obtained from the Final Safety Analysis Reports. Finally test results for a small LOCA will also be displayed and discussed.

A. Large LOCA

Figure 4A shows the containment response during a large LOCA (full line break) as given by the Final Safety Analysis Reports for the Crystal River Nuclear Power Station. The figure shows a plot of containment average pressure versus time after rupture.

The figure indicates that a peak containment pressure of approximately 49 psig (64 psia) is reached in slightly less than 20 seconds following the initiation of the LOCA. The value of this initial pressure peak depends on the amount of energy that is provided from the reactor coolant system (RCS) and the amount of energy that is transferred through the walls (condensation on the walls). Following a slight decrease in the containment pressure (3 psi), due to the action of the coolers, sprays, and the walls, a second rise in pressure occurs (49 psig). This second rise is a consequence of additional energy inputs from RCS. After the containment pressure peaks for the second time, the containment pressure will once more decrease slowly due to the actions of the coolers, sprays, and the walls.

Figure 4B shows the test results obtained after a large LOCA accident was initiated on the Crystal River simulator. The figure shows plots of average temperature, pressure, as well as liquid and vapour leak flows. A time frame of 5 minutes was chosen for this test. Note that the pressure versus time plot from the Final Safety Analysis Reports, converted to a linear scale, has been superimposed on figure 4B for comparison with the simulator results.

The test results show an initial pressure peak of 47 psig which is reached in less than 25 seconds from the initial LOCA. In the model, the proper pressure peak is achieved by tuning the overall heat transfer coefficient for the walls, which influences the amount of heat that is transferred through the walls. The initial peak is then followed by a brief period where the containment pressure decreases slightly (2 psi) until a second pressure peak occurs (48 psig). Following the second pressure peak, the containment pressure decays slowly due to the action of the sprays and coolers and walls.

The results show the successful implementation of first principles to correctly determine the pressure and temperature response in the containment following a large LOCA. This is clearly indicated by the close agreement between the test results, obtained from the Crystal River simulator, and the results given by the Final Safety Analysis Reports.

Figure 4A: RB Pressure vs Time (Large LOCA) - FSAR

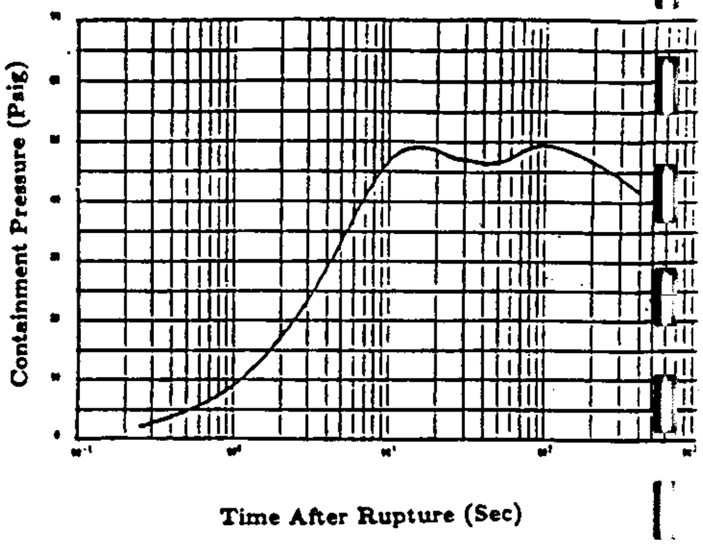
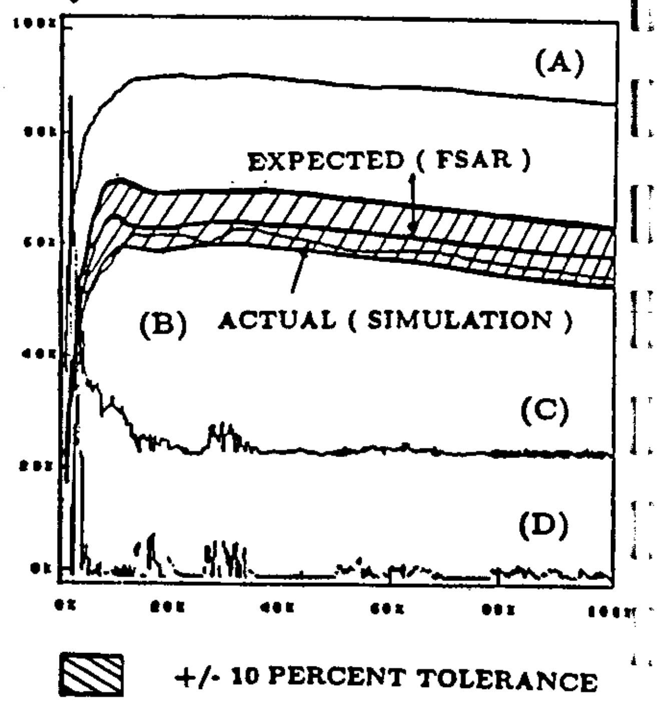


Figure 4B: Large LOCA Test Results - Crystal River Simulator



(A) Average Temperature (0 - 300 DegF)

(B) Average Pressure (0 - 100 Psia)

(C) LOCA Hot Leg Vapour Flow (-10000 - 35000 Lbm/s)

(D) LOCA Hot Leg Liquid Flow (0 - 90000 Lbm/s)

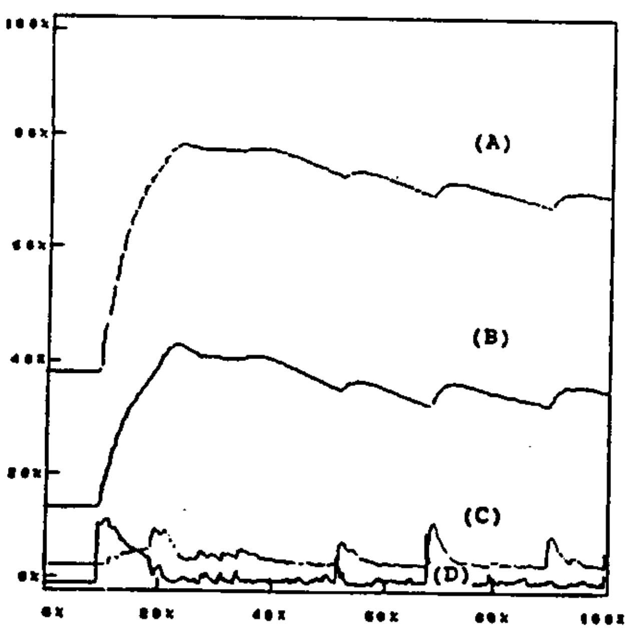
B. Small LOCA

Figure 5 displays the containment response during a small LOCA for the Crystal River simulator. The figure shows plots of average pressure, temperature, liquid and vapour leak flows versus time. A time frame of 10 minutes was chosen for this test.

During a small LOCA, the containment pressure does not climb to the values witnessed during a large LOCA. Consequently the containment average pressure does not reach the levels required to initiate the spray system. Therefore the only means available to decrease the containment temperature and pressure are the walls and the coolers.

Figure 5 shows the initial increase in pressure until a maximum value of approximately 40 psia is attained. Following this initial increase the containment pressure and temperature decrease slowly as a result of heat losses through the walls and the action of the coolers. Throughout the test, the pressure and temperature occasionally spike up. This is a result of intermittent bursts of energy from RCS into containment.

Figure 5: Small LOCA Test Results - Crystal River Simulator



(A) Average Temperature (0 - 300 DegF)

(B) Average Pressure (0 - 100 Psia)

(C) LOCA Hot Leg Vapour Flow (-1000 - 30000 Lbm/s)

(D) LOCA Hot Leg Liquid Flow (0 - 90000 Lbm/s)

Conclusion

A multiple node containment model was developed and proven to be successfully implemented on a number of simulators. The model was shown to be reliable in reproducing the proper containment responses during a wide range of transients. This is attributed to the application of first principles. In addition the test results indicate that evaporation from liquid spaces, flashing of leaks, condensation in coolers, on the walls, due to rain out, and due to the operation of the containment sprays are all reproduced faithfully.

The model is versatile. The techniques and methods used in developing this model can easily be implemented for any type of containment. The model is tailored to the specific requirements of the plant. The number of CV's and their location depend on the physical structures and restrictions, the equipment and their location, and the anticipated flow paths.

Finally, the model is easy to maintain and modify. This is due to the systematic approach used in its structure and mimimum effort required to fine tune the model to provide the required responses.

Acknowledgements

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Real Time Simulation of the Release and Transport of Radioactive Contaminants

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ABSTRACT

Calculating the responses of the radiation monitoring system (RMS) remains one of the most difficult aspects of nuclear power plant simulation to bring into the post-TMI, first principles simulator era. This task requires the simulation of the transport of radioactive contaminants, the transport of the radiation itself, and the instrument channel including the detector. The complex physics and lack of knowledge of input parameters have made these models lag the general simulator trend away from logical/heuristic modeling of physical systems. This paper describes a series of advances to the modeling methodology to change this situation.

The objective in the design of this model was to always calculate qualitatively reasonable radiation detector readings. This is the best that can be supported by the current level of scientific understanding of the phenomena involved. In particular, "the state-of-the-art of... (non-real-time research codes)... are accurate only within a factor of 100 even if all the accident conditions are known." (McKenna 1988). A secondary goal of the model design was to produce a real time model which includes all the important phenomena such that there would be no motivation to change the model until there is significant progress in the research code modeling in this field.

The purpose of the model is to allow realistic training of operators on the RMS responses. RMS is of major importance in the identification of the state of a nuclear power plant during many accidents. For operators to develop the skills needed to use this information, they need a simulator that includes the phenomena that caused "the radiation monitors during the TMI accident...(to be) more a source of 'puzzlement than of enlightenment'" (McKenna 1988).

MODEL SUMMARY

The model is divided into a fuel failure model, a transport model, and a detector model.

The fuel failure model consists of a mechanistic fuel failure calculation and a CINDER (England 1962) type calculation of the fission product inventory available to be released.

The transport model is a very fine mesh model, typically one radioactive contaminant node for every fluid model node. Transport is based on groups of isotopes which belong to the major chemical classes of radioactive contaminants of importance to nuclear power plants (noble gas, halogen, particulate, and N-16). A decentralized model is used, i.e., the transport calculations are performed in the simulator fluid model subroutines. A modularized approach to the transport equations allows the contaminant transport to be added as a separate module into an existing block structured model builder for fluid systems. Separate code fragments are introduced for rooms, for tanks, and for pipes. Tank nodes primarily allow partitioning of the various chemical classes between liquid and gas phases. Room nodes differ from tank nodes in that they allow the transport of aerosols and the plating out of contaminants on the walls of the room. The pipe nodes only transport. Phase change is not allowed within pipe nodes.

The detector model sums the contributions from the various sources of radiation, calculates its effect on the detector responses, and models the instrument channel. Sources include shine from possibly contaminated pipes and tanks, the "nameplate" radiation source, radiation carried in the room air, shine from normal operation radiation (such as gamma radiation from the N-16 in the primary coolant), radiation from materials which have plated out on the walls of rooms and equipment, direct gamma shine from the reactor, and the radiation from check sources and live zero sources. Shine is a somewhat unscientific term which refers to gamma radiation from "unexpected" sources. The "nameplate" radiation source is the process stream for which the detector was installed to monitor. Detector saturation, filter paper delay times, and statistical noise are all treated in this model.

Fuel Failure Model

The fuel failure mechanisms modeled are clad stress greater than the clad stress limit, clad strain greater than the clad strain limit, and the zirc-water reaction. Pellet cladding interaction (PCI) failures are currently only modeled as an instructor selected malfunction. The model requires a companion fuel rod model which calculates clad temperatures, fuel rod internal pressure, fuel pellet temperatures, and reactor core internal pressure. The zirc-water reaction thins the fuel rod cladding and contributes heat to the cladding temperature calculation in the fuel rod model. Fuel rod power census data from nuclear design code calculations are used to estimate detailed, rod by rod, power and temperature distributions where needed.

The power history effect on the nuclides available to be discharged to the coolant (the gap inventory) is calculated via a very simplified model of the diffusion of the fission products from the fuel pellet, but a fairly complete treatment of the radioactive decay chains. A CINDER-like calculation, with terms for the important fission product sources, their half lives, and their daughter products is performed in order to calculate the relative gap inventory of the different groups of isotopes modeled.

Transport Model

Four classes of radioactive isotopes are transported. Class 1 represents the halogens, transported primarily in the liquid phase. Class 2 represents the noble gases, transported primarily in the gaseous phase. Class 3 represents the particulates, transmitted in the liquid phase (and as aerosols). Each of these first three classes may have two groups of isotopes, one representing the short half life isotopes in the class and one representing the longer lived isotopes in the class. Class 4 represents N-16 and is calculated only in those fluid models which may see appreciable amounts of N-16 (half life of 7.1 seconds).

There are three kinds of transport nodes: pipe nodes, room nodes, and tank nodes.

Pipe nodes only transport. Pipe nodes do, however, transport all the groups. With two groups of isotopes modeled for each class, there will be six (or seven with N-16) concentrations of radioactive contaminants being transported with the fluid. Thus even though the noble gases are primarily present in gas spaces, they are also transported with the liquid stream in appropriate amounts. This is necessary due to the tiny amounts of radioactive isotopes which are observable on nuclear power plant RMS instrumentation.

Tanks are two region nodes having both a gas and a liquid space. Separate concentrations of each group of contaminants are calculated for each space. Tank nodes partition groups between the gaseous phase and the liquid phase. The noble gases are partitioned based on Henry's law data, with temperature as the primary independent variable. The halogens are partitioned based on a system dependent constant set such that the halogens are predominantly concentrated in the liquid space. The particulates are partitioned entirely into the liquid phase save for those situations (such as a pipe leak into a room with flashing to steam) in which they are transported as aerosols.

Rooms are different from tanks in that their surface area and volume are large as compared to the volume of water which may be expected to exist in the room. As such, the dominant mechanism is not the equilibrium between vapor and liquid spaces expected for tank nodes, but rather the processes of settling and plating out on the surfaces of the room.

For the case of a leak into a room, some of the water may flash into steam, depending on the enthalpy of the leaked fluid. The particulates associated with the mass of water which flashes to steam become an aerosol. The aerosol is modeled to gradually settle out of the room volume onto the room floor. Halogens are allowed to plate out onto all the surfaces of the room.

The model nodalization is determined in most cases by requiring a transport node for each fluid model node which may be isolated from other fluid model nodes. Exceptions include systems which cannot affect RMS detector readings. For instance, the model for the cooling water to the condenser may not need to include radiation transport.

In some models there are nodes which cannot be isolated one from another. For instance, the secondary side of a steam generator may typically have many nodes. These nodes would all be represented by a single tank model node in the transport model. In general, nodes which cannot be isolated from one another and which can be considered as being relatively well mixed with one another are combined for radiation transport purposes.

Detector Model

Two aspects of this model are particularly important to training as they can generate somewhat unexpected detector response. One is the radiation shine source to detectors which can cause detector response in many cases from other than the "nameplate" radiation source. Second is detector saturation, a phenomenon which can lead to a decrease in observed count rates while the actual level of radiation is greatly increasing.

Radiation shine is calculated to all radiation detectors from both global and local sources of radiation. Shine is based on local shine from the systems in the immediate vicinity of the detector, plus global shine from systems which have the capability to be large sources of radiation that will appreciably affect the readings on nearly all the radiation monitors in the plant. These sources shine to detectors based on the distance between the system and the detector and the attenuation due to the structural materials between the detector and each system. Local shine is based on 1) the identification of systems in the vicinity of each detector which have the capability of appreciable shine to the detector and on 2) an estimate of the attenuation caused by the local geometry and intervening material involved.

Detector count rates are adjusted for the effects of detector dead time. Detectors which have the detection of a count rate as a fundamental aspect of their operation will always have a dead time. Paralyzable detectors have the characteristic that a second event which occurs during the dead time following an event extends the dead period. This is not true of non-paralyzable detectors. Detectors which collect a current to calculate an integrated quantity of radiation do not have dead time.

Generally, then, the detector dead time characteristics are modeled as follows: GM tubes are paralyzable, scintillation counters are non-paralyzable, and ion chambers do not have dead time. But the entire instrument channel must be simulated to include the effects of electronics which may have been added to preclude observable saturation.

Non-paralyzable detector count rates are calculated from the following equation (Tsoulfanidis 1983):

$$g = \frac{n}{(1. + n\tau)}$$

where

g is the observed count rate

n is the actual count rate

t is the detector dead time.

One feature of this equation is that non-paralyzable detectors saturate at high count rates at a value equal to the inverse dead time.

Paralyzable detector count rates are calculated from the following equation (Evans 1955):

$$g = n \times exp(-n\tau)$$

This equation shows that the observed count rate can decrease due to an increase in the real count for paralyzable detectors at high count rates. This effect has been observed on the detectors during steam generator tube rupture transients (Nuclear Power Experience, 1975).

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CURIE TRANSPORT MODEL MODIFICATION ON TMI SIMULATOR

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ABSTRACT

The present TMI Curie Transport model consists of three portions: a source module, Primary side transport and BOP transport. The source module can be event-triggered or be triggered by high temperature and it consolidates all released radioactivities into four characteristic isotope groups. There are also four degrees of core damage and ten associated release categories which are determined by the Reactor Coolant System pressure and the in-core thermocouple temperatures. The transport model is embedded in the RCS model which simulates the nuclide conservation, decay and transport in the primary system flow paths. It also determines the nuclide transport to other balance-of-plant systems such as Makeup or Decay Heat Removal System and the leakage of the nuclide into the secondary system in the event of steam generator tube rupture. Also, transport of the nuclide in the secondary system is performed and the concentrations are converted to dose rate or other appropriate units for panel display.

The testing results have indicated that the modified model is capable of simulating the actual release, transport and response of the Radiation Monitoring System for steady state and transient operations. In addition, the simulator has also demonstrated that it is consistent with the NUREG-0737 guideline for initial estimate of the degree of core damage for severe accidents and the dose assessment calculations of the plant emergency preparedness model.

1. INTRODUCTION

In order to protect the public and the plant personnel against possible plant contamination, it is necessary to provide the monitoring of radiation levels in and around a nuclear power plant by placing radiation detecting devices in various vital areas. Naturally, the training for the proper operation of the Radiation Monitoring System (RMS) is imperative to the safety of the plant personnel as well as the safety of the surrounding general public. Therefore, it is essential to have a training simulator which is capable of simulating the phenomena and the response of the RMS during various circumstances such as steady state operation, iodine and cesium peaking after a rapid shutdown and release of large quantity of fission products into the primary system, secondary system and containment following a severe accident.

2. MODEL DESCRIPTION

The model consists of three major portions including source module, primary side curie transport module and BOP system transport module.

Source Module

Normal Condition: Under normal operating conditions, the radiation source normally come from the fission products of the fuel and the nuclear activation in the nuclear structure and its corrosive products(Sowden 1963). The radioactive fission products are normally confined in the fuel cladding. Some very trace amount of these isotopes may leak into the reactor coolant (Cohen 1980).

Abnormal Conditions: For abnormal conditions such as clad or core damage due to high temperature, the amount of activity release can increase the RCS coolant radiation level thousands of times higher than normal. The model uses NRC's NUREG 0737 Criterion 2(a) as the guideline regarding the initial estimate of the degree of reactor core damage.

The model permits 4 damage classes and 3 degrees of damage which result in 10 release categories as shown in Table 1 (Shoua 1983).

The categories or release are determined by the core pressure and temperature (in-core T/C's) curve as shown in Fig. 1 (B&W, 1970). Reactor Coolant System (RCS) pressure/temperature coordinates to the left of curve B represents "normal" RCS activity.

The region between curve B and curve C indicates potential for cladding failure and release to the RCS coolant of noble and volatile fission product gases contained in the gap formed by the fuel pellet and cladding. The curve D represents clad temperatures in excess of 2300 degree F. The region between curves C and D indicates the potential for fuel overheat and the release to the RCS coolant of noble and volatile fission product gases contained in the core fuel matrix.

The curve E in the Figure 1 represents cladding temperature in excess of 2800 degree F. It is assumed that in this region non-gaseous fission products contained in the fuel gap and fuel matrix will be released.

The regions between all curves will be broken down into three distinct sub-regions. Each sub-region will represent an NRC damage classification. The regions together with the NRC damage code classification and fraction released are also illustrated in Figure 1.

The category 1 is corresponding to NRC's Class 1, categories 2 through 4 are for class 2, categories 5-7 for class 3 and categories 8-10 for class 4.

Primary Side Curie Transport

The transport model is embedded in the RCS model which simulates the nuclide conservation, decay and transport in every fluid node of the primary system flow paths. Two additional activity balance equations are implemented to simulate rapid increase in the iodine and cesium activities following rapid power changes such as reactor trip (Makenna, 1988). It also determines the nuclide transport to other balance-of-plant systems such as Makeup or Decay Heat Removal System and the leakage of the radioactive nuclides into the secondary systems in the event of steam generator tube rupture which can be activated by Instructor Station through specific Malfunction tableau.

The modified model considers the following activity transport process:

- Transport of the activated material and fission products from fuel to the primary coolant (Makenna, 1988),

- Radiactive nuclide accumulation due to continuous release, and removal due to purification and decay (Makenna, 1988),

- Activity transport in water or in steam from primary side system to secondary system through steam generator tube rupture (Beahm, 1989).

BOP Curie Transport

The BOP systems that are to be considered in this stage of modification are limited to the pathways and components directly related to the radioactive material transport during a Steam Generator Tube Break. Conservation equations for radioactive material are implemented in every fluid node along main steam system to the condenser, and reactor building in case of Loss of Coolant Accident. Activity removal mechanism, including decay, dilution and being transported out of system, are also simulated.

3. PERFORMANCE TESTS AND RESULTS

Many tests are performed on the TMI plant reference simulator to verify the model performance against plant radiation monitor's readings.

An example test is discussed to demonstrate the model's capability of simulating rapid increase in the iodine and cesium activities following a reactor trip.

The total OTSG leakage rate is set at 0.03 GPM based on plant data. Before the reactor trip, the simulator was running at full power. Fig. 2 illustrate the behavior of radiation monitor's readings at letdown line taken from Recorder RM-L1.

Figure 2 identifies rapid increases in the iodine and total gamma activities as high as 1-2 orders of magnitude. This is in agreement with the plant data following a reactor trip on 11/29/87.

4. CONCLUSION

Radioactive material transport during normal operation or following a severe transient is a very complex process. In order to properly simulate the Radiation Monitoring System Response, extensive modifications of the current simulator system software are implemented on TMI plant reference simulator. The testing results have indicated that the present RMS model is capable of simulating the actual release, transport and response of the Radiation Monitoring System for steady state and transient operations. In addition, the simulator has also demonstrated that it is consistent with the NUREG-0737 guideline for initial estimate of the degree of core damage for severe accidents and the dose assessment calculations of the plant emergency preparedness model.

5. ACKNOWLEDGEMENT

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Table 1, Degree of Core Damage and Isotope Release

	Damage		Degree		
	clad	fuel	gap isotopes	fuel isotopes	rare earths
1	good	good	-	-	-
2	bad	good	10%	-	-
3	bad	good	50%	-	-
4	bad	good	100%	-	-
5	bad	overheat	100%	10%	-
6	bad	overheat	100%	50%	-
7	bad	overheat	100%	100%	-
8	bad	melt	100%	100%	10%
9	bad	melt	100%	100%	50%
10	bad	melt	100%	100%	100%

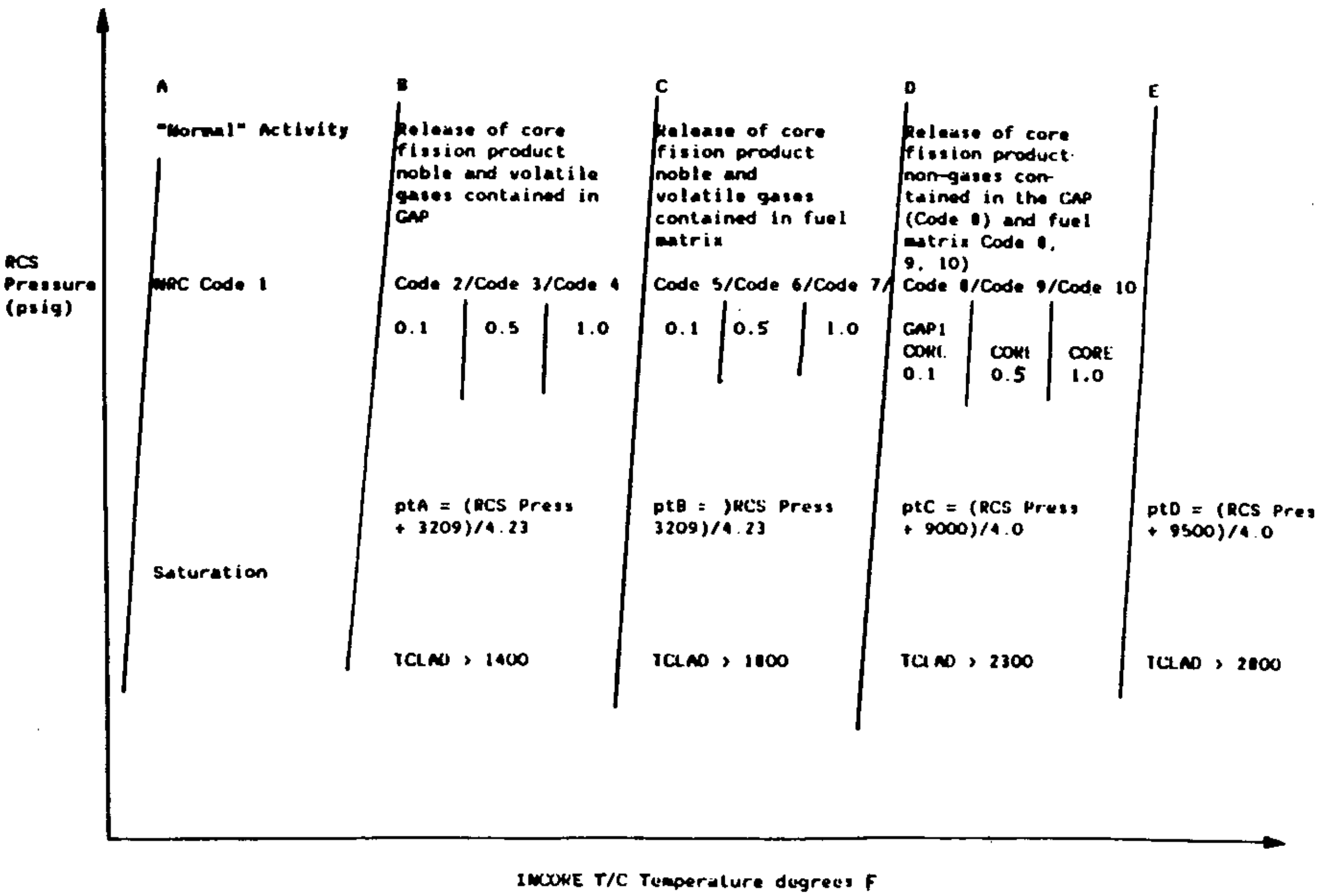

Figure 1, Degree of Core Damage Versus RCS pressure and Core Thermocouple or Cladding Temperature.

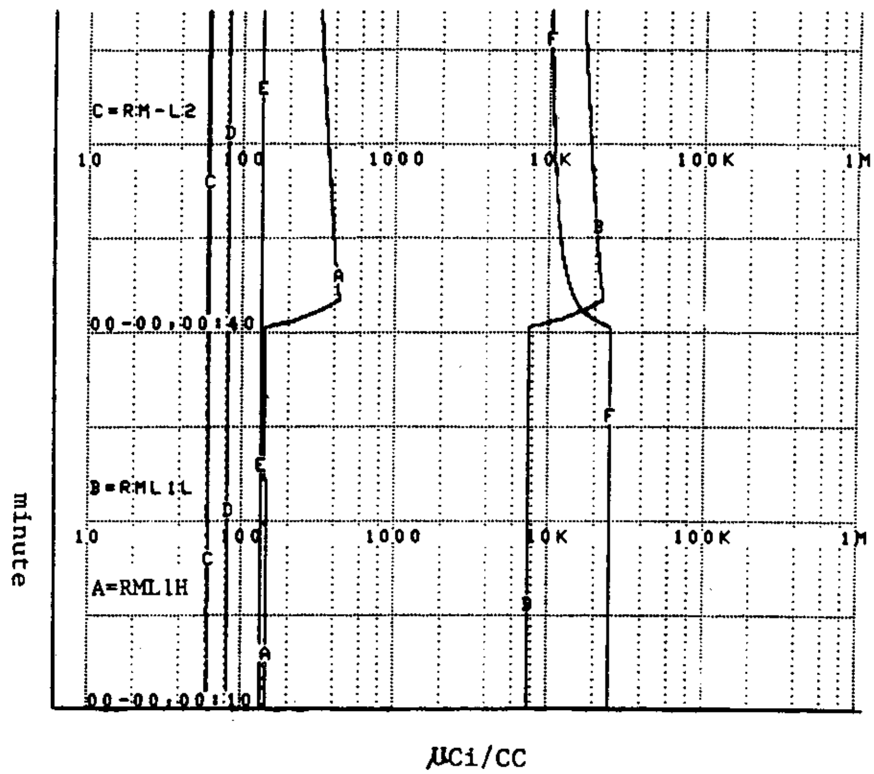

Figure 2, Radioactivity Spike Following a Reactor Trip

2.1 Momentum Equation

For a homogeneous mixture flowing in a constant cross-sectional area duct, the differential momentum equation can be written as follows:

$$\frac{\partial G}{\partial t} + \frac{\partial (G^2/\rho)}{\partial Z} = -\bar{g}\,\frac{\partial p}{\partial Z} - \frac{\partial F}{\partial Z} \qquad (1)$$

where

G = mass flux, $lb_m/(ft^2.s)$,

ρ = mixture density, lb_m/ft^3,

$\bar{g}$ = Conversion factor = 32.2 X 144,

p = pressure, psia

F = Fanning friction pressure drop,

and the gravity effects are neglected. For momentum consideration, the pressure is assumed to act at the upstream boundary of the nodes as shown in Figure (1).

Integrating Equation (1) from i to a and assuming uniform G and ρ within this interval, We obtain

$$\frac{dG_{ji}}{dt} = \frac{1}{L_{ji}} \left[\bar{g}\,(Pi - Pa) - Fia\right]$$

where

G_{ji} = mass flux in node i, $lb_m/(ft^2.s)$,

L_{ji} = length of node i, ft,

Pi = Pressure of node i, psia,

Fia = Fanning friction pressure drop between i and a.

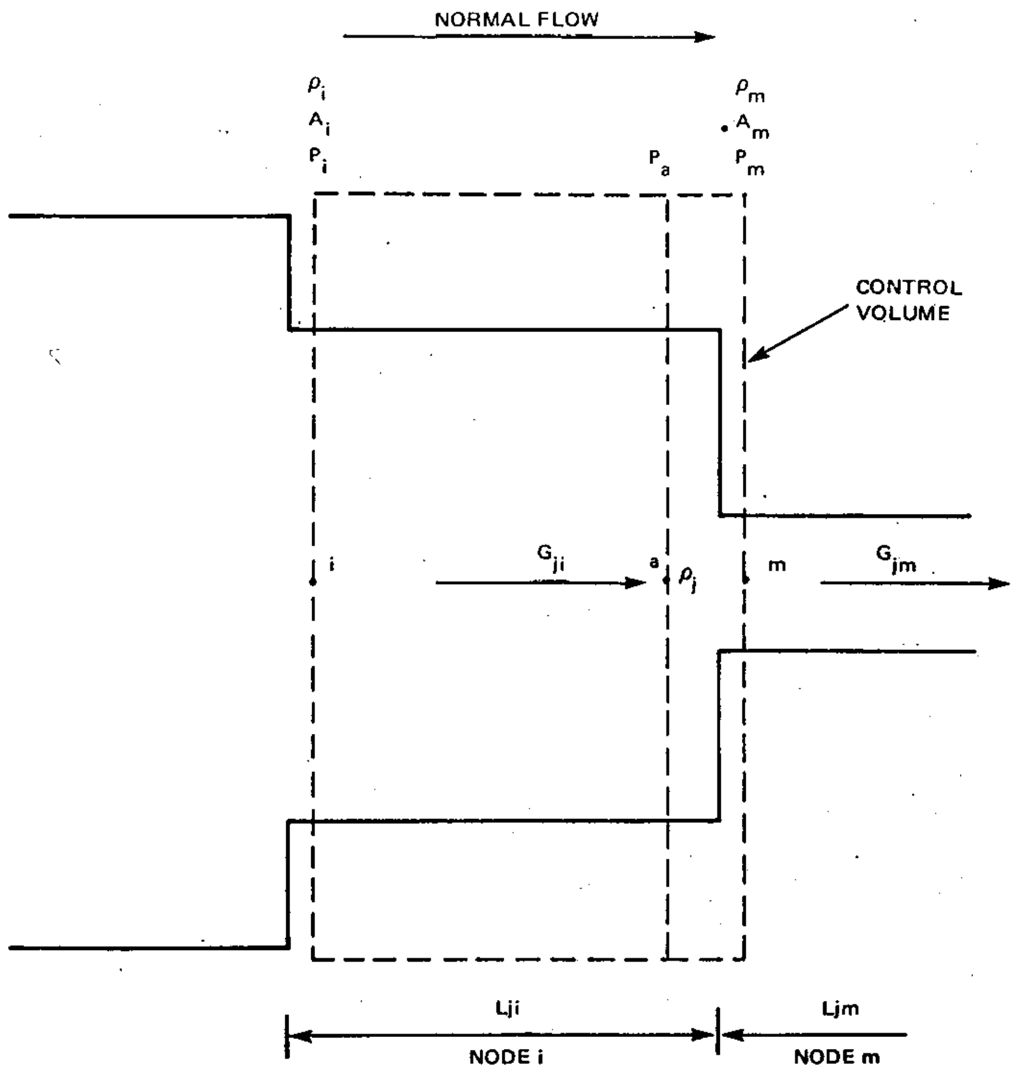

FIGURE 1 CONVENTIONS RELATED TO THE LINK-NODE GEOMETRY

In the region from a to m, the length is infinitesimally small. It is assumed that the fluid density denoted by ρ_j is constant across this abrupt area change and the expression commonly used for abrupt area change in single phase flow is applicable. That is,

$$(p_m - p_a)\,\overline{g} = (Kr - K)\,\frac{G_{ji}^{\;2}}{2\rho_j}$$

where p_m = pressure in node m, psia,

$Kr = 1 - \frac{A_i}{Am}^2$, coefficient which accounts for the reversible part of pressure drop,

A_i = flow area of node i, ft^2,

Am = flow area of node m, ft^2,

K = coefficient which accounts for the irreversible part of pressure drop.

Elimination of P_a from Equations (2) and (3) yields

$$\frac{dG_{ji}}{dt} = \frac{1}{L_{ji}} \left[\overline{g}\,(Pi - Pm) - Fia + (Kr - K)\,\frac{G_{ji}^{\;2}}{2\rho_j} \right] \tag{4}$$

Assuming

$$Fia = \left(f\,\frac{L}{D}\right)_{ji} \frac{|G_{ji}|}{2\rho_i}\,G_{ji}$$ (f = Fanning friction factor, D = hydraulic diameter)

$$\rho_i = \rho_j$$

and adding a term $(Kj_1 + Kj_2)\ |G_{ji}|G_{ji}/(2\ \rho i)$, to account for the loss other than those due to abrupt area changes between two adjacent nodes and Fanning friction, Equation (4) becomes

$$\frac{dG_{ji}}{dt} = \frac{1}{L_{ji}} \left\{ \bar{g}\ (Pi - Pm) - \left[(f\ \frac{L}{D})_{ji} + K_{j1} + K_{j2} \right] \frac{|G_{ji}|}{2\rho_i}\ G_{ji} + \left[\left[1 - (\frac{Ai}{Am})^2 \right] - K \right] \frac{G_{ji}{}^2}{2\rho_i} \right\} \tag{5}$$

$$= \Delta P$$

In the numerical integration of Equation (5), a first order explicit scheme is used. That is,

$$G_{ji}{}^{n+1} = G_{ji}{}^{n} + \Delta G_{ji} \tag{6}$$

where

$$\Delta G_{ji} = \Delta t \cdot \Delta P$$

n = index for time sequence,

Δt = time step, S.

The mass flux G_{ji}^{n+1} obtained from Equation (6) is then used to calculate the mass and energy transfers between node i and node m as described in the next section.

G. Radiological Safety, Hazard and Accident Analysis

Calculation of the internal and external dose rates, determination of reactor thermodynamic and hydrodynamic properties following an accident, e.g. release of radioactive materials, coolant system blow- down, steam generator rupture.

ACDOS3 ,N ACTIVATION ACTIVITIES & DOSE RATES	CCC-0442	03 89
ACFA ,ISOTOPE ACTIVATION OF COOLANT & STRUC MAT	NEA 1072	12 87
ACRO ,ORGAN DOSES FROM ACUTE OR CHRONIC RADIOACTIVE INHALATION OR INGESTION	NEA 0621	11 81
ACT-ARA ,TIME-DEP RADN SOURCE TERMS	CCC-0372	03 93
AERIN ,ORGAN & TISSUE DOSES FROM RADIOACTIVE AEROSOLS	NESC0908	12 89
AFS ,ADVANCED TEST REACTOR ALARM FILTER SYS	NESC9584	01 88
AIRDIF ,N & GAMMA DOSES FROM NUCL EXPLOSION BY 2-D AIR DIFFN	CCC-0360	05 85
AIRDOS-PC ,CLEAN AIR ACT COMPLIANCE SOFTWARE FOR PC	CCC-0551	11 90
AIREKMOD-RR ,REACTIVITY TRANSIENTS IN NUCL RESEARCH REACTORS	IAEA1274	09 93
AIRGAMMA ,EXTERNAL GAMMA-RAY EXPOSURE FROM RADIOACTIVE CLOUD	NEA 1130	10 89
AIRSCAT ,DOSE RATE FROM GAMMA AIR SCAT BY SINGLE SCAT APPROXN	CCC-0341	07 82
AIRTRANS ,TIME-DEP,ENERGY DEP 3-D N TRANSP,GAMMA TRANSP IN AIR BY M-C	CCC-0110	07 82
ALARM-B1 ,THERMOHYDRAULICS OF BWR WITH JET PUMPS DURING LOCA	NEA 0500	05 85
ALARM-P1 ,PWR THERMOHYDRAULICS FOR ECCS DURING BLOWDOWN	NEA 0705	04 83
ALDOSE ,DOSE RATE FROM ALPHA DISK SOURCE IN H2O	CCC-0577	12 95
AMRAW ,RISK ASSESSMENT METH FOR RADIOACTIVE WASTE MANAGEMENT	USCD0795	09 81
ANVENT ,TEMP DISTR & PRESS IN CONTAINMENT & ICE CONDENSER AFTER LOCA FOR LWR	NESC0529	04 79
APUD-3.0 ,OFF-SITE CONTAMINATION ASSESSMENT FROM ACCIDENTAL RELEASE	IAEA1219	10 93
ARIANNA-2 ,SUB-COMPARTMENT THERMO-HYDRAULIC TRANSIENTS IN LOCA	NEA 1368	10 93
ARLEKIN ,GENERAL POINT REACTOR KINETICS BY LIE-SERIES METH	NEA 0174	04 81
ASCOT-1 ,THERMOHYDRAULICS OF AXISYM PWR CORE WITH HOMOG FLOW DURING LOCA	NEA 0539	06 81
AT123D ,1-D,2-D,3-D TRANSIENT WASTE TRANSP SIMULN IN GROUNDWATER	CCC-0417	04 83
ATR ,RADN TRANSP MODELS IN ATMOSPHERE AT VARIOUS ALTITUDES	CCC-0179	07 82
AVACOM-ETAP ,AVAILABILITY & ELEMENT TRANSIENT & ASYMPTOTIC REPAIR PROCESS	NEA 1076	03 87
AWE-1 AWE-2 BRUNA ,MINIMAL CUT SETS OF LOGIC TREES	NEA 0861	04 83
BACFIRE ,MINIMAL CUT SETS COMMON CAUSE FAILURE FAULT TREE ANAL	NESC1020	08 84
BEACON/MOD3 ,1-D & 2-D 2 PHASE FLOW & HEAT TRANSF IN CONTAINMENT,LWR LOCA	NESC0767	05 83
BLAST ,ACCIDENT COND IN CRIT & SUBCRIT THERM REACTOR SYS	IAEA0820	04 81
BLOW-3A ,NA BOILING IN LMFBR ACCIDENT	NEA 0943	03 84
BLT ,WASTE TRANSP THROUGH POROUS MEDIA FROM CONTAINER FAILURE	CCC-0633	06 97

BRA ,BREAST RADN ANAL FROM MAMMOGRAPHY	IAEA0915	11 93
BRUCH-D-06 ,LOCA OF PWR PRIM SYS WITH 23 CONTROL VOL & 9 RUPTURE POINTS	NEA 0418	11 92
BURST ,TIME-DEP PRESS & COOLANT FLOW AFTER CIRCUIT FRACTURE IN HTGR	NEA 0114	12 80
BURST-1 ,RUPTURE OF 1-D CYL PRESSURIZED LIQUID SYS,HYDRODYN CALC	NESC0435	04 79
BWR-GALE ,RADIOACTIVE GASEOUS & LIQUID WASTE RELEASE FROM BWR	NESC1080	03 88
BWR-LTAS ,BWR LONG TERM ACCIDENT SIMULN PROG	CCC-0485	04 87
CAAC ,SYS TO IMPLEMENT ATMOSPHERIC DISPERSION ASSESSMENTS	CCC-0476	06 86
CAMERA CAM ,RADN DOSE ABSORPTION BY COMPUTER MAN	CCC-0240	12 94
CANSWEL-2 ,CLADDING CREEP DEFORMATION OF PWR DURING LOCA	NEA 0885	03 84
CAP-88 ,DOSE RISK ASSESSMENT FROM AIR EMISSIONS OF RADIONUCLIDES	CCC-0542	03 94
CARNAC ,N FLUX & N SPECTRA IN CRITLTY ACCIDENT	NEA 0393	06 81
CASSANDRE ,2-D REACTOR DYN FEM PROG WITH THERMOHYDRAULIC FEEDBACK	NEA 0712	06 87
CAT ,CORE HEAT TRANSIENTS IN LOCA	NEA 0054	12 80
CAUCASE ,AUTOMATIC CHARACTERIZATION OF NEUTRONIC FLUCTUATIONS SPECTRUM	NEA 1092	06 86
CBLR ,AIR MASS VARIATION IN NUCL POWER PLANT CONTAINMENT BUILDING	NEA 0947	02 84
CEBUG ,3-D TRANSIENT HYDRAULICS FOR NA H2O REACTN BY FINITE ELEM METH	NESC0548	04 79
CHARGE-2/C ,FLUX & DOSE BEHIND SHIELD FROM ELECTRON,P,HEAVY PARTICLE IRRADN	CCC-0070	12 94
CHART ,1-D,3-D I DECAY HEAT THERM ANAL OF CONTAINMENT AFTER ACCIDENTS	NESC0804	11 80
CHEMLOC-2 ,CORE HEATING & METAL STEAM REACTN AFTER LOCA & ECCS FAILURE	NESC0366	11 80
CLAPTRAP ,PRESS TRANSIENTS IN LWR CONTAINMENT DURING LOCA	NEA 0631	06 81
COBRA ,TRANSIENT THERMOHYDRAULICS FUEL ELEM CLUSTERS,SUBCHANNEL ANAL METH	NESC0432	05 89
COBRA-3C/RERTR ,THERMOHYDRAULIC LOW PRESS SUBCHANNEL TRANSIENTS ANAL	NESC9978	02 92
CODAR ,RADIATION DOSES FROM RADIOACTIVE RELEASES AROUND BRITISH ISLES COASTS	NEA 0940	03 84
COMCAN ,FAULT TREE ANAL,MINIMAL CUT SETS FOR COMMON CAUSE FAILURE	NESC0704	01 86
COMIDA ,RADIONUCLIDE FOOD CHAIN MODEL FOR ACUTE FALLOUT DEPOSITION	PSR-0343	11 94
COMPBRN3 ,MODELLING OF NUCL POWER PLANT COMPARTMENT FIRES	ESTS0023	10 94
COMRADEX-4 ,DOSES FROM RADIOACTIVE RELEASE,METEOROLOGICAL DISPERSION,AEROSOL	NESC0663	04 79
CONDOS-II ,RADN DOSE FROM CONSUMER PRODUCT DISTR CHAIN	CCC-0416	09 84
CONTEMPT ,LWR CONTAINMENT PRESS & TEMP DISTR IN LOCA	NESC0433	02 90
CONTEMPT-4MOD3 ,LWR CONTAINMENT LONG-TERM PRESS DISTR & TEMP DISTR IN LOCA	NESC0818	06 92
CORAN ,PWR & BWR CONTAINMENT RESPONSE TO LOCA	NEA 0567	04 81
CORCON-MOD2 ,MOLTEN-CORE CONCRETE INTERACTIONS	NESC1096	03 89
CORRAL-2 ,FP RELEASE & DEPOSITION IN LWR CONTAINMENT,MELTDOWN ACCIDENT	NESC0745	12 88
COSTANZA-RZ ,1-D LIQUID COOLED REACTOR DYN IN R-Z GEOM	NEA 0333	04 81
COSTAX-BOIL ,TRANSIENT DYN ANAL OF BWR & PWR IN AX GEOM	NEA 0398	06 81
COSTAX-BWR ,COUPLED TIME-DEP 2 GRP N DIFFN & 2 PHASE FUEL ROD COOLANT FLOW	NEA 0533	02 81

CRAC2 ,REACTOR ACCIDENT RISK ASSESSMENT	CCC-0419	01 89
CRRIS ,HEALTH RISK ASSESSMENT FROM ATMOSPHERIC RELEASES OF RADIONUCLIDES	CCC-0518	03 90
DEIS ,IMPACT MEASURES OF LOW LEVEL RADIOACTIVE WASTE DISPOSAL	CCC-0455	06 86
DELFIC ,TES,GAMMA DOSES FROM NUCL EXPLOSION RADIOACTIVE CLOUDS	CCC-0257	05 85
DENZ ,DENSE TOXIC OR EXPLOSIVE GASE DISPERSION IN ATMOSPHERE	NEA 0840	07 86
DEPCO-MULTI ,SUBCOOLED DECOMPRESSION IN PWR PRIM SYS LOCA	NEA 0453	06 81
DISDOS ,KERMA IN MODEL MAN FROM EXTERNAL GAMMA SRCE	CCC-0170	11 81
DISKTRAN ,DETECTOR RESPONSE TO SCALAR FLUX FROM DOT-4 CALC	CCC-0533	04 89
DISPER-1 ,AEROSOL PARTICLE LEAKAGE & RELEASES FROM CONTAINMENT	NESC0554	11 80
DISPERS ,RADIOACTIVE RELEASE INTO SURFACE WATER & GROUND WATER	CCC-0454	07 86
DORIAN ,BAYES METH PLANT AGE RISK ANAL	NESC1146	02 92
DOSE-SGTR ,IODINE RELEASE DURING SGTR IN PWR	CCC-0624	11 95
DOSEFACTOR-DOE ,DOSE RATE CONV FACTORS FOR PHOTON & ELECTRON EXPOSURE	CCC-0536	12 93
DOSET ,HTGR TIME-DEP FP RELEASE ACCIDENT DOSES,DECAY CHAIN CALC	NESC0261	04 79
DRUCK ,THERM,MECH STRESS OF PWR FUEL ROD DURING LOCA BLOWDOWN	IAEA0885	07 86
DRUCKSCHALE-44 ,PRESS & TEMP TRANSIENTS IN BLOWDOWN ACCIDENT	NEA 0215	04 81
DRUFAN-01/MOD2 ,TRANSIENT THERMOHYDRAULICS OF PWR PRIM SYS LOCA	NEA 0839	04 83
DRUGEVO ,TIME-DEP CONTAINMENT PRESS & TEMP IN BWR OR PWR LOCA	NEA 0457	02 81
DSNP ,PROG & DATA LIB SYS FOR DYN SIMULN OF NUCL POWER PLANT	NESC0784	07 88
DUST-BNL ,RADIOACTIVE WASTE TRANSP FROM CONTAINER LEAKS INTO GROUND WATER	CCC-0634	08 96
DWNWND ,DOWNWIND ATMOSPHERIC CONCENTRN & DISPERSION BY GAUSSIAN PLUME MODEL	CCC-0383	04 83
EDISTR ,NUCL DATA BASE GEN FOR INTERNAL RADN DOSIMETRY CALC	PSR-0191	09 85
EDO ,DOSES TO MAN & ORGANS FROM REACTOR OP NOBLE GAS & LIQUID WASTE RELEASE	NEA 0845	10 83
EGAD ,GROUND LEVEL GAMMA DOSES FUNC OF GAMMA ENERGY FOR RADIOACTIVE RELEASES	NESC0600	05 83
ELBA ,BREMSSTRAHLUNG DOSE FROM ISOTR ELECTRON FLUX ON PL AL SHIELD	CCC-0119	07 82
EMERALD ,RADN RELEASE & DOSE AFTER PWR ACCIDENT FOR DESIGN ANAL & OP ANAL	NESC0546	07 88
EMERALD-NORMAL ,ROUTINE RADN RELEASE & DOSE FOR PWR DESIGN ANAL & OP ANAL	NESC0685	07 78
ESDORA ,CONTINUOUS & INSTANTANEOUS FP RELEASE INTO ATMOSPHERE	NEA 0561	06 81
EUREKA ,REACTIVITY TRANSIENTS IN LWR FROM CONTROL ROD,COOLANT FLOW,TEMP	NEA 0447	06 81
EVNTRE ,NUCL POWER PLANT PROBABILISTIC ACCIDENT RISK ANAL EVENT TREE	NESC1132	01 91
EXCURS ,HEAT TRANSF TRANSIENTS IN CYL REACTOR CHANNEL LOCA	NEA 0424	06 81
EXCURS-3 ,REACTOR KINETICS & HEAT TRANSF IN CYL CHANNEL DURING ACCIDENT	NEA 0228	04 81
EXCURS-3-RR ,KINETICS OF RESEARCH REACTOR REACTIVITY TRANSIENT ANAL	IAEA1273	09 93
FAPCO ,EVALN OF FLAWS IN NUCL POWER PLANT COMPONENT STRUC	IAEA0868	02 84
FASTGRASS ,GASEOUS FP RELEASE IN UO2 FUEL	NESC1095	09 90
FIRAC ,NUCL POWER PLANT FIRE ACCIDENT MODEL	NESC1092	10 89

FIRENET ,PROBABILISTIC RISK ASSESSMENT OF FIRES IN BUILDINGS	NEA 1042	10 86
FLASH4 FLASH6 ,PRESS TRANSIENTS SIMULN IN PWR & BWR LOCA	NESC0448	10 86
FLODIS ,THERM RESPONSE OF FSV HTGR CORE	NESC9597	12 87
FONTA ,RADN RELEASE IN ATMOSPHERE & DEPOSITION IN HUMAN ORGANS	NEA 0669	04 83
FRANTIC-NRC ,ACCIDENT SEQUENCE & EVENT TREE ANAL FOR SYS AVAILABILITY & OP	NESC0766	01 88
FRAP-T ,TEMP & PRESS IN OXIDE FUEL DURING LWR LOCA	NESC0658	11 84
FRAPCON2 ,STEADY-STATE LWR OXIDE FUEL ELEM BEHAV,FP GAS RELEASE,ERROR ANAL	NESC0694	02 88
FREADM-1 ,REACTOR KINETICS THERMOHYDRAULICS CALC FOR FAST REACTOR ACCIDENTS	NESC0479	03 72
FRELIB ,FAILURE RELIABILITY INDEX CALC	NEA 0692	08 84
FRETA-B ,LWR FUEL ROD BUNDLE BEHAV DURING LOCA	NEA 0982	04 92
FTA ,FAULT TREE ANAL FOR MINIMAL CUT SETS,GRAPHICS FOR CALCOMP	NESC0666	11 80
FTAP ,MINIMAL CUT SETS OF ARBITRARY FAULT TREES	NESC9659	11 86
GASPAR ,DOSES FROM RADIOACTIVE RELEASES & METEOROLOGICAL DATA,COST BENEFIT	NESC0963	04 91
GASPAR-II ,RADN EXPOSURE TO MAN FROM AIR RELEASES OF REACTOR EFFLUENTS	CCC-0463	02 93
GENII ,ENVIRONMENMTAL RADIATION DOSIMETRY SYS	NESC9465	03 92
GENII-1.485 ,ENVIRONMENTAL RADIATION DOSIMETRY SYS	CCC-0601	05 97
GRASS-SST ,FP GAS RELEASE & FUEL SWELLING IN STEADY-STATE & TRANSIENTS	ESTS0075	08 95
HAA3B ,HETEROG AEROSOL TRANSP AFTER LMFBR ACCIDENTS,LOGNORMAL SIZE DISTR	NESC0443	11 80
HAARM ,TIME-DEP DIFFN & DEPOSITION OF RADIOACTIVE AEROSOLS,LMFBR ACCIDENTS	NESC0797	02 84
HADOC ,EXTERNAL & INTERNAL ORGAN DOSES FROM RADN RELEASE AT HANFORD	CCC-0452	08 87
HAMOC ,PRESS TRANSIENTS IN REACTOR VESSEL PIPING SYS AFTER ACCIDENTS	NESC0710	05 79
HARAD ,DECAY ISOTOPE CONCENTRN FROM ATMOSPHERIC NOBLE-GAS RELEASE	CCC-0387	06 86
HERALD ,WIDE-AREA RADN DOSES TO POPULATION FROM LOCA	IAEA0938	01 90
HEXANN-EVALU ,N IRRADN OF REACTOR PRESS VESSELS	NEA 1125	09 89
HGSYSTEM ,ATMOSPHERIC DISPERSION FOR IDEAL GASES & HF	ESTS0545	03 97
HOPE ,INTEGRAL ANAL OF LMFBR TRANSIENT OVERPOWER ACCIDENT	NEA 0618	04 81
HORN ,FP TRANSP IN PRIM COOLANT SYS OF BWR & PWR IN LOCA	NEA 1169	02 90
HOTSPOT ,FIELD EVALN OF RADN RELEASE FROM NUCL ACCIDENT	CCC-0644	03 97
HYDY-B1 ,CHANNEL THERMOHYDRAULICS DURING LOCA OF BWR,PWR	NEA 0499	06 81
HYTRAN ,OPEN CHANNEL THERM & HYDRAULIC TRANSIENTS IN LOCA	NEA 0216	04 81
IMPACTS-BRC2.1 ,GEN RADIOLOGICAL IMPACTS ANAL	ESTS0005	08 94
IMPAIR-2 ,I DIFFN FROM ACCIDENTAL FP RELEASE IN AQUEOUS & GASEOUS SPECIES	NEA 1274	04 93
IMPORTANCE ,MINIMAL CUT SETS & SYS AVAILABILITY FROM FAULT TREE ANAL	NESC0779	08 84
INFLTB ,DOSIMETRIC MASS ENERGY TRANSFER & ABSORPTION COEFF	PSR-0313	03 97
INREM-EXREM-3 ,TIME-DEP ORGAN DOSES FROM ISOTOPE INHALATION & INGESTION	CCC-0185	07 82
INTEG INSPEC ,ACCIDENT FREQUENCIES & SAFETY ANAL FOR NUCL POWER PLANT	NESC0590	04 79

INTERTRAN ,RADN EXPOSURE FROM VEHICLE TRANSPORT OF RADIOACTIVE MAT	IAEA0886	04 86
IRDAM ,INTERACTIVE RAPID DOSE ASSESSMENT FROM REACTOR ACCIDENT EFFLUENTS	ESTS0109	08 96
IRRAS ,INTEGRATED RELIABILITY & RISK ANAL SYS FOR PC	ESTS0003	08 95
KAOS-V ,N FLUENCE TO KERMA FACTOR EVAL FROM ENDF/B-5 & JENDL-2	PSR-0306	09 91
KENO-4-VP ,VECTORIZATION OF CRITLTY SAFETY PROG KENO-4	NEA 1319	06 90
KINE ,1-D PWR DYN WITH PARTIAL CORE BOILING	NEA 1002	07 85
KRONIC ,ANNUAL BODY TISSUE DOSE FROM CONTINUOUS ATMOSPHERIC RELEASE	CCC-0229	05 83
LAVA/CS ,COMPUTER SECURITY RISK ASSESSMENT	NESC9579	06 88
LEPRICON ,PWR VESSEL DOSE ANAL WITH DORT & ANISN PROG	PSR-0277	04 92
LISA ,HAZARD ASSESSMENT OF NUCL WASTE DISPOSAL IN GEOLOGICAL FORMATIONS	NEA 0860	02 92
LOCA-MARK-2 ,FUEL TEMP & CLAD TEMP IN HWR STEAM GENERATOR LOCA	NEA 0623	06 81
LPGS ,RADN EXPOSURE FROM RADIOACTIVE RELEASE INTO HYDROSPHERE	CCC-0385	07 84
LPSC ,P & N FLUX,SPECTRA BEHIND SLAB SHIELD FROM P IRRADN	CCC-0064	07 82
LSHINSE ,AIR SCAT N & GAMMA DOSERATES FOR COMPLEX SHIELDING GEOM	NEA 1306	02 91
LSL-M2 ,N SPECTRA LOG ADJUSTMENT FOR DOSIMETRY APPL	PSR-0233	07 95
MABEL-2 ,CLADDING DEFORMATION IN PWR FUEL RODS DURING LOCA	NEA 0911	03 84
MACCS1.5 ,RADIOACTIVE ATMOSPHERIC RELEASE IN REACTOR ACCIDENTS	ESTS0002	10 94
MANYCASK ,RADN DOSE RATE AROUND MANY CASKS	NEA 1047	01 94
MARCH ,CONTAINMENT BEHAV AFTER LOCA,BLOWDOWN,MELTDOWN,METAL H2O REACTN	NESC0734	02 93
MARINRAD ,HEALTH HAZARD FROM RADIOACTIVE MAT RELEASE INTO OCEAN	CCC-0503	12 87
MATRA ,VOID SIMULN IN STEAM & H2O MIX CHANNEL IN ACCIDENT	NEA 0380	01 81
MAVRAC ,DOSE DISTR IN ASTRONAUT MODEL & SPACE VEHICLE	CCC-0023	11 81
MELT-3 ,THERMOHYDRAULICS & NEUTRONICS,FAST REACTOR TRANSIENTS WITH FEEDBACK	NESC0700	07 78
MESORAD ,EMERGENCY RESPONSE AIRBORNE DOSE ASSESSMENT	ESTS0331	03 95
MILDOS-AREA ,RADIOLOGICAL IMPACT OF AIRBORNE U238 FROM MINING & MILLING	NESC9460	05 90
MIRDOSE ,INTERNAL RADN DOSE USING MIRD SCHEMA	NESC9623	12 87
MKENO-DAR ,CRITLTY REACTOR SAFETY ANAL BY M-C WITH DIRECT ANGL REPRESENTATION	NEA 0996	03 97
MOCARS ,SYS RELIABILITY & MINIMAL CUT SETS USING M-C METH	NESC0912	05 83
MOCUS ,MINIMAL CUT SETS & MINIMAL PATH SETS FROM FAULT TREE ANAL	NESC0653	11 80
MTR PC ,MODULAR SYS FOR NEUTRONICS,THERMOHYDRAULIC & SHIELDING CALC	IAEA1336	07 96
MUCHA1 ,FUEL ROD PAIR THERMOHYDRAULICS DURING LOCA & ECCSA FOR LWR	NESC0508	12 72
MULTI-KENO ,CRITLTY SAFETY ANAL BY M-C	NEA 0933	11 93
MULTIPLET ,LARGE EVENT TREES FOR RISK ASSESSMENT CALC	NEA 1041	08 88
MUP ,UNCERTAINTY PROPAGATION FOR NUCL SAFETY ANAL BY M-C	NEA 0991	10 86
MUTUAL ,NUCL REACTOR CRITLTY SAFETY ANAL FOR ARRAY SYS	NEA 1176	05 88
NAC ,N ACTIVATION ANAL & ISOTOPE INVENTORY	CCC-0164	02 95

NAHAMMER ,PRESS TRANSIENTS IN NA LMFBR PIPING SYS,LIN FLUID HAMMER THEORY	NESC0717	05 79
NAIAD ,LOCA TRANSIENT & STEADY-STATE 2 PHASE FLOW IN CHANNEL NETWORK	NEA 0806	05 81
NALAP ,THERMOHYDRAULICS FOR NA COOLED LMFBR AFTER PIPE RUPTURE & ACCIDENTS	NESC0780	11 80
NATRAN-2 ,LMFBR PIPING SYS PRESS TRANSIENTS,FLUID HAMMER & NA H2O REACTN	NESC0719	05 79
NATRANSIENT ,LMFBR PIPING SYS PRESS TRANSIENTS,FLUID HAMMER,NA H2O REACTN	NESC0718	05 79
NAUA-MOD5 NAUA-MOD5/M ,AEROSOLS IN REACTOR CONTAINMENT DURING MELTDOWN	NEA 0853	11 88
NFCLIST ,RADIONUCLIDE DECAY DATA TABULATION	ESTS0352	10 94
NOABL ,3D FREE DIVERGENCE WINDFIELD CALCULATION	NEA 1534	06 97
NORCOOL ,BWR LOCA ANAL WITH THERM NON-EQUILIBRIUM & COUNTER CURRENT FLOW	NEA 0671	04 83
NOTAM ,NEUTRONICS HYDRAULICS OF BWR IN STEADY-STATE COND	NEA 0921	02 90
NUTRAN ,DOSES BY RADIONUCLIDE MIGRATION FROM NUCL WASTE STORAGE	NESC9888	08 88
OCA-P ,PWR VESSEL PROBABILISTIC FRACTURE MECHANICS	NESC1125	06 90
OCTAVIA ,PWR PRESS VESSEL FAILURE PROBABILITY FOR ROUTINE PRESS TRANSIENTS	NESC0898	05 84
ORION-II ,CONCENTRN & DOSE FROM RADIOACTIVE RELEASE INTO ATMOSPHERE	NEA 1249	12 90
PARET ,THERMOHYDRAULICS OF REACTIVITY ACCIDENT IN LWR	NESC0555	05 94
PART61 ,LOW LEVEL RADIOACTIVE WASTE IMPACT ANAL	CCC-0499	09 90
PAVAN ,ATMOSPHERIC DISPERSION OF RADIOACTIVE RELEASES FROM NUCL POWER PLANTS	CCC-0445	03 91
PC-PRAISE ,BWR PIPING RELIABILITY ANAL	ESTS0071	06 97
PCAUS ,PROBABILITY OF RADIOGENIC CANCER	NESC9605	07 87
PCDOSE-ESTSC ,RADIOACTIVE DOSE ASSESSMENT & NRC VERIFICATION	ESTS0764	06 96
PHAMISS ,TIME-DEP SYS RELIABILITY FOR PHASED MISSIONS	NEA 1182	04 88
PHAZE ,COMPONENT FAILURE RATE STATISTICAL ASSESSMENT	NESC1147	02 92
PIEDEC ,EFF DOSE EQUIVALENT FROM INHALATION OR INGESTION	NEA 1084	10 89
PIFITE ,CRBR HCDA COVER GAS PIPE LINE ANAL	NESC9790	01 86
PL-MOD ,FAULT TREE ANAL FOR OP SYS USING A MODULARIZING METH	NESC0897	05 83
PLUDOS ,GROUND LEVEL GAMMA DOSE FROM RADIOACTIVE RELEASE AT VARIOUS HEIGHTS	NEA 0493	06 81
PLUMEX ,GAMMA DOSES FROM ATMOSPHERIC PLUME	NEA 0704	03 93
POPDOS ,INTEGRAL DOSES IN ENGLAND & WALES FROM ATMOSPHERIC RADN RELEASE	NEA 0653	06 81
PRASMA ,RISK ANAL OF OFF-SITE PROTECTION FROM REACTOR ACCIDENTS	NEA 1352	04 91
PRECIP-2 ,ZIRCALOY CLADDING OXIDATION SIMULN FOR LWR UNDER LOCA COND	NEA 0904	05 83
PREP KITT ,SYS RELIABILITY BY FAULT TREE ANAL	NESC0528	11 80
PREST ,PRESS TEMP TRANSIENTS,I INHALATION IN CONTAINMENT BUILDING FROM LOCA	NEA 0251	04 81
PRISIM ,PROBABILISTIC RISK ASSESSMENT SYS FOR NRC NUCL PLANTS	NESC1126	02 92
PROCIV ,PROTECTION COEFF FROM FALLOUT IN RESIDENTIAL AREA HOUSING	NEA 0695	10 83
PROSA-1 PROSA-2 ,ACCIDENTS PROBABILITY ANAL USING RESPONSE SURFACE METH	NESC0778	02 84
PSAPACK ,PROBABILISTIC SAFETY ANAL WITH FAULT EVENT TREES	IAEA1174	09 89

PTA2 ,PRESS TRANSIENTS OF LARGE PIPING NETWORKS IN LMFBR	NESC0761	08 84
PTH-1 ,PRESS & TEMP IN CONTAINMENT AFTER BLOWDOWN OF H2O COOLANT SYS	NESC0155	11 80
PWR-GALE ,RADIOACTIVE GASEOUS RELEASE & LIQUID RELEASE FROM PWR	NESC1081	03 88
QAD-P5 ,FAST N & GAMMA FLUX,DOSE RATE,ENERGY DEPOSITION IN SHIELD	CCC-0307	04 84
QBF ,RADN DOSE DISTR AROUND SPENT FUEL SHIPPING CASKS	CCC-0617	04 96
QUINCE ,DOSE ABSORPTION,HEALTH RISK FROM SKIN CONTAMINATION	CCC-0556	10 94
RABFIN PARTS ,NOBLE GAS,IODINE,PARTICULATE GASEOUS EFFLUENT DOSE PARAM	ESTS0062	12 95
RACC ,BIOLOGICAL HAZARD POTENTIAL OF FUSION SYS	CCC-0388	07 82
RADRISK ,DOSES TO HUMAN ORGANS & HEALTH EFFECTS FROM INHALATION & INGESTION	CCC-0422	12 84
RADTRAN3 ,RADIOACTIVE MAT TRANSP RISK	CCC-0508	11 94
RAS ,FAULT TREE ANAL,RELIABILITY,MINIMAL CUT SETS FOR COMMON CAUSE FAILURE	NESC0889	06 82
RASCAL-2.0A ,RADIOLOGICAL DOSES FROM ACCIDENTAL RELEASE TO ATMOSPHERE	CCC-0553	12 95
RASCAL1.3 ,RADIOLOGICAL DOSE ASSESSMENT FROM ACCIDENTAL RELASE TO ATMOSPHERE	NESC1127	08 91
RATAF ,RADIOACTIVE LIQUID TANK FAILURE	ESTS0050	08 94
RBD ,DOSES FROM RADIONUCL INHALATION,INGESTION,WOUND UPTAKE FROM BIOASSAYS	CCC-0632	12 95
RCSLK9 ,PWR COOLANT SYS LEAK RATE	NESC1090	10 88
REBEL-3 ,WHOLE BODY & ORGAN GAMMA DOSES OF INHOMOG PHANTOM BY M-C	IAEA0846	07 86
RELAP-3B/M110 ,FLOW TEMP PRESS STEAM QUALITY IN LWR AFTER LOCA & ACCIDENTS	NESC0733	11 80
RELAP-4 ,TRANSIENT 2 PHASE FLOW THERMOHYDRAULICS,LWR LOCA & REFLOOD	NESC0369	07 85
RELAP-5 ,TRANSIENT 2 PHASE FLOW THERMOHYDRAULICS,LWR LOCA ACCIDENTS	NESC0917	08 85
RELAP-UK ,THERMOHYDRAULIC TRANSIENTS & STEADY-STATE OF LWR	NEA 0437	04 83
RELAP/REFLA ,CORE REFLOODING DURING PWR LOCA	NEA 0821	04 83
RELOSS ,RELIABILITY OF SAFETY SYS BY FAULT TREE ANAL	NEA 0615	03 85
REMIT ,RADNEXPOSURE MONITORING & INFORMATION TRANSMITTAL SYS	ESTS0579	03 97
REMO ,FAILURE ANAL OF SYS WITH REPARABLE & STANDBY COMPONENTS BY M-C	NEA 0429	06 81
RESRAD ,RESIDUAL RADIOACTIVE MAT GUIDELINE IMPLEMENTATION	CCC-0552	03 97
RETRAC ,REACTOR CORE ACCIDENT SIMULN	IAEA1286	12 94
RIBD ,FP INVENTORY & DELAY HEAT IN FAST REACTORS,WITH DATA LIB,IB36	CCC-0137	12 89
RISKAP ,RISK ASSESSMENT OF RADN EXPOSURE FOR POPULATION	CCC-0486	08 92
RISKIND ,RADIOLOGICAL RISK ASSESSMENT FOR SPENT NUCL FUEL TRANSP	CCC-0623	11 94
RSAC ,GAMMA DOSES,INHALATION & INGESTION DOSES,FP INVENTORY AFTER FP RELEASE	NESC0265	04 79
SAFE ,FAIL-TO-SAFE,FAIL-TO-DANGER ANAL OF PROTECTIVE NETWORKS	ESTS0446	07 94
SAFTAC ,M-C FAULT TREE SIMULN FOR SYS DESIGN PERFORMANCE & OPTIMIZATION	NESC0674	11 80
SALP-3 ,FAULT TREE ANAL BY LIST PROCESSING METH	NEA 0863	08 87
SAS-1A ,LMFBR TRANSIENT ANAL WITH HEAT TRANSF,FUEL DEFORMATION & FEEDBACK	NESC0400	04 79
SCALE-3 ,MODULAR SYS FOR CRITLTY,SHIELDING,HEAT TRANSF CALC	CCC-0466	12 87

TERFOC-N ,RADIATION DOSES IN FOOD CHAIN FROM ATMOSPHERIC RELEASE	NEA 1328	09 91
THACT-RR ,ANAL OF THERMAL HYDRAULICS TRANSIENTS IN RESEARCH REACTOR CORE	IAEA1272	09 93
THALES ,THERMOHYDRAULIC LOCA ANAL OF BWR & PWR	NEA 0774	01 90
THETA-1B ,FUEL ROD TEMP DISTR BY 2-D DIFFN,HEAT TRANSF TO COOLANT,LWR LOCA	NESC0512	04 79
THYDE-B2 ,THERMOHYDRAULIC TRANSIENTS DURING LOCA OF BWR	NEA 0778	03 89
THYDE-P ,PWR LOCA THERMOHYDRAULIC TRANSIENT ANAL	NEA 0779	02 92
TIBSO ,NUCL TRANSITIONS & RADIOACTIVITY MIGRATION IN TECHNOLOGICAL SYS	IAEA0945	05 95
TINA ,BLOWDOWN PHASE ANAL OF LOCA IN PWR OR BWR	NEA 0699	04 81
TIRION-4 ,ATMOSPHERIC DISPERSION OF RADIOACTIVE MAT FOR VARIOUS WEATHER COND	NEA 0701	12 82
TORAC ,FLOWS,PRESS,MAT TRANSP WITHIN STRUC DURING TORNADO	NESC1093	01 90
TOXRISK ,TOXIC GAS RELEASE ACCIDENT ANAL	NESC9710	02 93
TRAC ,THERMOHYDRAULICS,REACTOR KINETICS,2 PHASE FLOW LOCA ANAL	NESC0836	05 94
TRAC-BD1 ,LOCA ANAL OF BWR WITH 3-D PRESS VESSEL & MULTI BUNDLE FUEL MODEL	NESC1031	12 84
TRACO-V ,PRESS & TEMP TRANSIENTS IN PWR CONTAINMENTS DURING LOCA	IAEA1236	09 91
TRANS-ACE ,RADIOACTIVE MAT TRANSP IN REPROCESSING PLANT FIRE ACCIDENT	NEA 1291	01 90
TRANS-FUGUE-1 ,SINGLE CHANNEL 2 PHASE FLOW HEAT TRANSF AFTER BOILING	NESC0268	11 80
TRANSV2 ,LOCA & STEADY-STATE THERMOHYDRAULIC ANAL OF MTR	IAEA1209	02 93
TRAPSCO-2 ,PRESS & TEMP TRANSIENTS IN PWR SUBCOMPARTMENTS DURING LOCA	NEA 0745	02 84
TRIPM ,ISOTHERM TRANSP & DECAY OF RADIONUCLIDES IN AQUIFER	NESC1028	05 85
UDAD ,RADN EXPOSURE TO MAN AT U PROCESSING PLANT	NESC0824	10 83
UMIBIO ,U MILL BIOASSAY DOSIMETRY MODEL	NESC1088	01 88
VARSKIN-2 ,DOSE CALC FOR SKIN CONTAMINATION,WITH SADDE INPUT GENERATOR	CCC-0522	08 95
VISA-2 ,REACTOR VESSEL FAILURE PROBABILITY UNDER THERM SHOCK	NESC1115	05 90
VPI-NECM ,NUCL ENGINEERING PROG COLLECTION FOR COLLEGE TRAINING	IAEA0871	07 91
WEERIE ,RADIOACTIVE RELEASE FROM REACTOR TO COOLING CIRCUIT & ATMOSPHERE	NEA 0610	09 81
WHAM-6 ,PRESS & VELOCITY TRANSIENTS IN FLUID PIPES,WAVE SUPERPOSITION METH	NESC0278	07 91
WRAITH ,INTERNAL & EXTERNAL DOSES FROM ATMOSPHERIC RELEASE OF ISOTOPES	CCC-0427	05 84
WREM TWODEE-2/MOD3 ,2-D TIME-DEP FUEL ELEM THERM ANAL AFTER PWR LOCA	NESC0712	05 83
ZOCO-6 ,TEMP TRANSIENTS IN BWR & PWR CONTAINMENT DURING LOCA	NEA 0401	06 81
ZYLIND ,GAMMA PENETRATION FOR CYL SOURCE & SHIELD GEOM	NEA 1251	02 93
ZZ ACTL82 ,DATA LIB OF EVALUATED ACTIVATION X-SEC	DLC-0069	03 97
ZZ AMPX-2/123 ,123 GRP N X-SEC LIB FROM ENDF/B-4 BY PROG AMPX-2	NEA 0886	03 97
ZZ BROND ,EVALUATED N DATA LIB IN ENDF/B-5 FMT	IAEA0949	03 97
ZZ BWRSB-RINGHALS1 ,STABILITY BENCHMARK DATA FROM BWR RINGHALS-1	NEA 1454	06 95
ZZ DECAYREM/C ,DECAY SPECTRA LIB FOR EXREM CALC	DLC-0030	03 97
ZZ DLC-10B AVKER ,N KERMA RESPONSE FUNC DATA LIB	DLC-0010	03 97

ZZ DLC-11 RITTS ,121 GRP COUPLED X-SEC FOR ANISN,DOT,MORSE	DLC-0011	03 97
ZZ DLC-14 AIR ,GRP CONST LIB OF SECONDARY GAMMA TRANSP IN AIR FOR ANISN CALC	DLC-0014	03 97
ZZ DLC-17 NOX ,119 GRP COUPLED X-SEC OF NITROGEN,O,AIR FOR MORSE	DLC-0017	03 97
ZZ DOSCOV ,24 GRP COVARIANCE DATA LIB FROM ENDF/B-5 FOR DOSIMETRY CALC	DLC-0090	03 97
ZZ DOSDAT-2 ,GAMMA & ELECTRON DOSE CONV FACTOR DATA LIB FOR BODY ORGANS	DLC-0079	03 97
ZZ DOSEDAT-DOE ,DOSERATE CONV FACTORS FOR EXTERNAL PHOTON,ELECTRON EXPOSURE	DLC-0144	03 97
ZZ DRALIST ,RADIOACTIVE DECAY DATA FOR DOSIMETRY & HAZARD ASSESSMENT	DLC-0080	03 97
ZZ FGR-DOSE ,DOSE COEFF FOR INTAKE & EXPOSURE TO RADIONUCLIDES	DLC-0167	03 97
ZZ FUELS-DATA ,DATA LIB FOR LWR FUEL BEHAV FOR FRAP PROG	NESC0844	11 85
ZZ IRDF-82 ,620 GRP X-SEC LIB & SPECTRA FOR DOSIMETRY CALC IN ENDF/B-5 FMT	IAEA0867	03 97
ZZ LAHIMACK ,MULTIGRP N & GAMMA X-SEC & RESPONSE FUNC UP TO 800 MEV	DLC-0128	03 97
ZZ MGCL-26/137 ,137 GRP CONST LIB FROM ENDFB-4 FOR CRITLTY SAFETY ANAL	NEA 0936	03 97
ZZ NUCDECAY ,NUCL DECAY DATA FOR RADN DOSIMETRY CALC FOR ICRP & MIRD	DLC-0172	03 97
ZZ RADDECAY ,DECAY DATA LIB FOR RADIOLOGICAL ASSESSMENT	DLC-0134	03 97
ZZ RECOIL/B ,HEAVY CHARGED PARTICLE RECOIL SPECTRA LIB FOR RADN DAMAGE CALC	DLC-0055	03 97
ZZ SKYPORT ,IMPORTANCE FUNC FOR N & GAMMA FOR SKYSHINE DOSE FROM ACCELERATOR	DLC-0093	03 97
ZZ SKYSDATA-KSU ,N & GAMMA SKYSHINE RESPONSES	DLC-0188	03 97
ZZ SNLRML ,DOSIMETRY X-SEC RECOMMENDATIONS	DLC-0178	03 97
ZZ UNGER ,EFFECTIVE DOSE EQUIVALENT DATA FOR SELECTED ISOTOPES	DLC-0164	03 97

See Authorization information

NESC0433 CONTEMPT,LWR CONTAINMENT PRESS & TEMP DISTR IN LOCA 900223

1. NAME OR DESIGNATION OF PROGRAM - CONTEMPT-LT/028, CONTEMPT-LT/026.

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED

Program-name	Package-ID	Orig. Computer	Test Computer
CONTEMPT-CONPS	NESC0433/02	IBM 360 series	IBM 360 series
CONTEMPT-LT	NESC0433/03	IBM 370 series	IBM 370 series
CONTEMPT-LT/28-H	NESC0433/08	IBM 3090	DEC VAX 8810
CONTEMPT-LT026	NESC0433/04	IBM 370 series	IBM 370 series
CONTEMPT-LT26B	NESC0433/01	IBM 370 series	IBM 370 series
CONTEMPT-LT28B	NESC0433/06	CDC 7600	CDC 7600
CONTEMPT-LT28B	NESC0433/07	CDC 7600	CDC 7600
CONTEMPT-PS	NESC0433/05	IBM 360 series	IBM 360 series

3. DESCRIPTION OF PROBLEM OR FUNCTION

CONTEMPT-LT was developed to predict the long-term behavior of water-cooled nuclear reactor containment systems subjected to postulated loss-of-coolant accident (LOCA) conditions. CONTEMPT-LT calculates the time variation of compartment pressures, temperatures, mass and energy inventories, heat structure temperature distributions, and energy exchange with adjacent compartments. The program is capable of describing the effects of leakage on containment response. Models are provided for fan cooler and cooling spray engineered safety systems. One to four compartments can be modeled, and any compartment except the reactor system may have both a liquid pool region and an air-vapor atmosphere region above the pool. Each region is assumed to have a uniform temperature, but the temperatures of the two regions may be different. The user determines the compartments to be used, specifies input mass and energy additions, defines heat structure and leakage systems, and prescribes the time advancement and output control. CONTEMPT-LT/28-H (NESC0433/08) includes also models for hydrogen combustion.

4. METHOD OF SOLUTION

The initial conditions of the containment atmosphere are calculated from input values, and the initial temperature distributions through the containment structures are determined from the steady-state solution of the heat conduction equations. A time advancement proceeds as follows. The input water and energy rates are evaluated at the midpoint of a time interval and added to the containment system. Pressure suppression, spray system effects, and fan cooler effects are calculated using conditions at the beginning of a time-step. Leakage and heat losses or gains, extrapolated from the last time-step, are added to the containment system. Containment volume pressure and temperature are estimated by solving the mass, volume, and energy balance equations. Using these results as boundary conditions, the heat conduction equations describing structure behavior are advanced using an implicit technique. The resulting heat transfer rates are used to correct the previous estimates of the water and energy storage in the containment volume, and the containment conditions are obtained by solving for the second time the containment balance equations. The pressure suppression routines use the conditions at the beginning of a time-step to calculate both the initial explusion of water from the vents and the flow through the vents. From the calculated flow rates, mass and energy are removed from the dry well and added to the wet well.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM

Maxima of -
- 20 heat conducting structures
- 101 mesh points for each structure
- 20 regions for each structure
- 50 flow elements in one segment of the horizontal vent pressure suppression system
- 10 horizontal vents (or branches) in a segment
- 50 reductions within an input time-step

CONTEMPT-LT can be used for analyzing the transient containment behavior of boiling-water reactors (BWRs) including Mark I, Mark II, and Mark III systems; pressurized-water reactors (PWRs), and experimental water reactor simulators or related experiments.

6. TYPICAL RUNNING TIME

On the CDC 7600, CONTEMPT-LT/028 requires less than 30 seconds to run the two sample problems. On the IBM 360/75, CONTEMPT-LT/026 requires approximately 0.021 second per time advancement with 90 mesh points for heat structures without pressure suppression. The pressure suppression timing is not easily predicted but run time for two sample problems ranges from 0.3 to 2 seconds per time advancement.
CONTEMPT/LT28B (NESC0433/06): NEA-DB executed the test case on CDC 7600 in 70 seconds.

7. UNUSUAL FEATURES OF THE PROGRAM

8. RELATED AND AUXILIARY PROGRAMS

CONTEMPT-LT/028 is the most recent of a series of computer programs developed to describe the thermal-hydraulic conditions attendant to Various postulated transients in the containment of light-water reactor systems. CONTEMPT-LT/026 replaced CONTEMPT (NESC Abstract 297), CONTEMPT-CONPS, CONTEMPT-PS, CONTEMPT-LT/022, and CONTEMPT-LT/025 (which contained known errors and was never distributed by NESC).

9. STATUS

NESC0433/02	: Arrived at NEADB
	: in preparation
	: Obsolete
NESC0433/03	: Arrived at NEADB

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                : in preparation
                : Obsolete
NESC0433/08     : Requested by NEADB
                : Arrived at NEADB
                : in preparation
                : Screened
NESC0433/04     : Arrived at NEADB
                : Obsolete
NESC0433/01     : Arrived at NEADB
                : Tested at NEADB
                : Obsolete
NESC0433/06     : Requested by NEADB
                : Arrived at NEADB
                : in preparation
                : Tested at NEADB
                : Obsolete
NESC0433/07     : Requested by NEADB
                : Arrived at NEADB
                : Screened
NESC0433/05     : Arrived at NEADB
                : Obsolete
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NESC0433/06 :
- Don W. Hargroves et al.:
  CONTEMPT-LT/028 - A Computer Program for Predicting Containment Pressure-Temperature Response to a Loss-of-Coolant Accident.
  NUREG/CR-0255 TREE-1279 R4 (March 1979).
- NESC Note 81-30 (December 18, 1980).

NESC0433/07 :
- NESC Note 81-30 (December 18, 1980)
- Don W. Hargroves et al.:
  CONTEMPT-LT/028 - A Computer Program for Predicting Containment Pressure-Temperature Response to a Loss-of-Coolant Accident.
  NUREG/CR-0255 TREE-1279 R4 (March 1979).

NESC0433/05 :

11. MACHINE REQUIREMENTS - CONTEMPT-LT26

430 kbytes and a CALCOMP or SC4060 plotter for graphical output.
CONTEMPT/LT28B (NESC0433/06): To execute the sample case on CDC 7600 storage requirements are: 160,000 octal words (SCM)
100,000 octal words (LCM).
plotter (CONTEMPT-LT/029); 430K bytes and a Calcomp or SC4060 plotter for graphical output (CONTEMPT-LT/026).

12. PROGRAMMING LANGUAGE USED

NESC0433/02	: FORTRAN+ASSEMBLER
NESC0433/03	: FORTRAN-IV
NESC0433/08	: FORTRAN+ASSEMBLER
NESC0433/04	: FORTRAN-IV
NESC0433/01	: FORTRAN+ASSEMBLER
NESC0433/06	: FORTRAN-IV
NESC0433/07	: FORTRAN-IV
NESC0433/05	: FORTRAN-IV

13. OPERATING SYSTEM UNDER WHICH PROGRAM IS EXECUTED

SCOPE 2.1 (CDC 7600), OS/360 MVT (IBM360), MVS/XA (IBM).

14. OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS

The thermodynamic properties of water and steam required by the program are generated by the STH20 program and made available as a library data set.

15. NAME AND ESTABLISHMENT OF AUTHOR - Contributed by

CDC 7600	D.W. Hargroves, L.J. Metcalfe, and Teh-Chin Cheng* EG&G Idaho, Inc. P.O. Box 1625 Idaho Falls, Idaho 83415
IBM 360	L.L. Wheat and W.J. Mings Idaho National Engineering Laboratory EG&G Idaho, Inc. P.O. Box 1625 Idaho Falls, Idaho 83415
CONTEMPT-LT/28H	E.N.E.L. Thermal and Nuclear Research Center PISA (Italy)

* Contact

16. MATERIAL AVAILABLE

NESC0433/02 :

Report:

INFORMATION	0 records
SOURCE PROG.+DD CARDS+SAMPLE PROBLEMS	0 records
PRINTED OUTPUT OF THE FIRST SAMPLE PROB.	0 records

NESC0433/03 :

Report:

SOURCE PROGRAM (F4)	0 records
ENVIRONMENTAL SUBROUTINES (F4)	0 records
ENVIRONMENTAL SUBROUTINES (ASS)	0 records
PLOTTING ROUTINE PLAS (F4)	0 records
OVERLAY CARDS	0 records
SAMPLE PROBLEM DATA	0 records
SAMPLE PROBLEM PRINTED OUTPUT	0 records

NESC0433/08 :
CONTEMPT-LT/28-H Information File
FORTRAN Routines
ASSEMBLER Routines
CONTEMPT Binary Steam Table
JCL to create Load Module
JCL to run Sample Problem
S. P. Input CASE1 (BWR) without Hydrogen
S. P. Input CASE2 (PWR) without Hydrogen
S. P. Input CASE3 (BWR) with Hydrogen
S. P. Input CASE4 (PWR) with Hydrogen
S. P. Output CASE1 (BWR) without Hydrogen
S. P. Output CASE2 (PWR) without Hydrogen
S. P. Output CASE3 (BWR) with Hydrogen
S. P. Output CASE4 (PWR) with Hydrogen
Report: NUREG/CR-0255 TREE-1279 R4 (March 1979)
Report: MTC/FL53/86 (April 1986) (in Italian)
Report: MTC/SW40/84 (June 1984) (in Italian)
NESC Note 81-30 (December 18, 1980)

Information file	55 records
CONTEMPT-LT/28-H FORTRAN source	15788 records
CONTEMPT-LT/28-H ASSEMBLER source	1209 records
Steam tables (binary for IBM)	0 records
JCL to create load module	26 records

See Authorization information

NEA 1368 ARIANNA -2,SUB-COMPARTMENT THERMO-HYDRAULIC TRANSIENTS IN LOCA 931020

1. NAME OR DESIGNATION OF PROGRAM - ARIANNA-2.

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED

Program-name	Package-ID	Orig. Computer	Test Computer
ARIANNA-2	NEA 1368/01	IBM 30xx series	DEC VAX 6000

3. DESCRIPTION OF PROGRAM OR FUNCTION

ARIANNA-2 allows to analyze the behaviour of a thermal-hydraulic transient following a LOCA in a multicompartment containment system during short, medium and long term accidental sequences. The transient is described as a quasi-steady state: mass and energy flows, in each time step, are based on the thermodynamic conditions of the previous time step.. The mass and energy inventory in each volume may be modified by the contribution of heat transfer, junction flows and blow-down mass and energy inputs. The flow through the junctions can be evaluated by choosing one of the following three models: Moody, homogeneous inertial flow or orifice polytropic flow. Each control volume can be simulated as a homogeneous steam-water-air mixture or as a stagnant mixture region (atmosphere) above a liquid pool (sumps); the pool region may or may not be in thermodynamic equilibrium with the atmosphere. In the blow-down volumes, isentropic or isenthalpic expansion of the mixture jet is possible together with associated de-entrainment of liquid in the water pool.

4. METHOD OF SOLUTION

Mass and energy exchange between volumes, linked through junctions, are calculated according to the following assumptions:
- inlet and outlet pressures for each junction are equal to those of the two linked volumes;
- thermodynamic flow conditions are equal to those of the donor volume;
- flow properties are constant during the time-step and mixture flow is homogeneous.

The inertial flow is calculated on the basis of a numerical solution of the momentum equation while the orifice flow is calculated on the basis of Moody model (critical case) or ideal gas model. The heat transfer to structures is calculated solving the Fourier equation by a finite difference method; and various options make it possible to calculate heat transfer coefficients.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM

The ARIANNA-2 program provides up to 100 volumes, 200 junctions, 30 of which can have time-dependent areas, and 100 heat conducting structures which can exchange with any combination of compartments or between any compartment and the outside atmosphere. Up to 101 mesh-points for each structure are allowed and up to 20 different material regions.

6. TYPICAL RUNNING TIME

A 2-volumes, 1-junction problem requires 0.005 s per time step of CPU time on IBM 3090.
NEA 1368/01: NEA-DB compiled the source program and executed the two test cases included in this package on a DEC VAX-6000 computer in 5m10.54s and 18.36s, respectively.

7. UNUSUAL FEATURES OF THE PROGRAM

The architecture of ARIANNA-2 is completely modular. Additional features such as restart, plotter file automatic time step control permit flexible use of available code features.

8. RELATED AND AUXILIARY PROGRAMS

The ARIANNA-2 is the last in the ARIANNA series of programs originally developed at DCMN of Pisa University.

9. STATUS

NEA 1368/01 : Requested by NEADB
: Arrived at NEADB
: in preparation
: Tested at NEADB

10. REFERENCES

NEA 1368/01 :
- N. Cerullo, F. Oriolo and A. Pasculli:
  ARIANNA-2 , Un Nuovo Codice di Calcolo per l'Analisi di Transitori Termoidraulici in Sistemi di Contenimento a Piena Pressione (in Italian)
  RL 098(84).
- M. Cascioli, N. Cerullo, W. Flospergher, F. Oriolo and S. Paci:
  Implementazione e Qualifica dei Modelli di Salvaguardia Ingegneristica del Contenimento nel Codice ARIANNA-2 (in Italian)
  RL 305(87).
- N. Cerullo, A. Mandrefini, F. Oriolo and S. Paci:
  Validation of the ARIANNA-2 Code on the Basis of HDR V44 and T31.5 Tests
  Second International Conference on Containment Design and Operation
  Toronto (October 14/17, 1990).
- N. Cerullo, W. Flospergher, F. Oriolo and A. Pacsulli:
  A Model for Thermal-Hydraulic Analysis of Transients in PWR Containment Systems the ARIANNA-2 Computer Code
  Reprint from Energia Nucleare/Anno 2/N. 3 (December 1985).
- NEA Data Bank:
  Graphical Comparison of the Original IBM Results (full line) with those Obtained on the DEC VAX 6000 (marks)

NEADB (19/10/93).

11. MACHINE REQUIREMENTS

4 MB of memory are needed for compilation and execution on IBM 3090.

12. PROGRAMMING LANGUAGE USED

NEA 1368/01 : FORTRAN-IV

13. OPERATING SYSTEM UNDER WHICH PROGRAM IS EXECUTED

MSV, VM (IBM 370).
NEA 1368/01: VAX/VMS V5.5-2 with VAX Fortran-77 compiler.

14. OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS

Numerical output may show system-dependent variations.

15. NAME AND ESTABLISHMENT OF AUTHORS

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Telefax 39-50-585265, Telex 500104 FINGPI I

16. MATERIAL AVAILABLE

NEA 1368/01 :
HDR Input File
HDR Output File
DCMN Input File
DCMN Output File
ARIANNA Source Program
Report: RL 098(84) (in Italian)
Report: RL 305(87) (In Italian)
DCMN 013(90) (October 14/17, 1990)
Reprint from Energia Nucl. (December 1985)
NEADB (19/10/93)

ARIANNA-2 information file	190 records
ARIANNA-2 source code	3764 records
ARIANNA-2 job control for installation	29 records
ARIANNA-2 test input file # 1	110 records
ARIANNA-2 test output file # 1	1892 records
ARIANNA-2 test input file # 2	34 records
ARIANNA-2 test output file # 2	2002 records

17. CATEGORY = G , H ,