Development of a Multi-dimensional two Phase Thermal Hydraulic Analysis Code, MEDUSA, and its application to Feed Line Break Analysis

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1. Introduction

A general purpose multi-dimensional two phase thermal hydraulic analysis code, MEDUSA has been developed utilizing two fluid three field governing equations. The flexible noding system of MEDUSA allows users to model the wide variety of geometry encountered in nuclear reactor system. The specific models of reactor kinetics, pump, valves, and separators has been also developed to simulate the core power and various hydraulic components of primary or secondary systems of nuclear power plant. In this paper, the main feature of the MEDUSA code is briefly described, and its application results to feed line break(FLB) accident of Korean Standard Nuclear Power Plant(KSNP) are compared with those of the licensing analysis code, CESEC-III.

2. Main feature of MEDUSA

2.1 Governing equations and solution scheme

The governing equations of the MEDUSA code consist of the following two fluid three field equations.

- Continuity equations for liquid, droplet, vapor, and non-condensable gas
- Momentum equations for liquid, droplet, and the vapor/non-condensable gas mixture
- Energy equations for liquid/droplet and vapor/non-condensable gas which are assumed to be respectively at the thermal equilibrium state

Moreover, three dimensional conduction equation is used to solve the radial, axial, azimuthal temperature distribution within heated structures including nuclear fuel rod, and one dimensional conduction equation is used for unheated structures. The core neutronic response is simulated using point kinetics model, and the core decay heat is calculated with 1971 or 1979 ANS models. The conservation equations for each of the three fields are solved using a semi-implicit, finite-difference numerical technique.

2.2 Constitutive equations

Specific correlations for the interfacial heat transfer to super heated vapor, subcooled vapor, superheated liquid, and subcooled liquid are given at various flow regimes such as bubbly flow, liquid film flow, inverted annular flow, Chunk flow, droplet flow, etc. Interfacial area is also based on flow regime, but in the dispersed droplet flow regime, it is determined from the interfacial area concentration transport equation. Also, the entrainment and de-entrainment model is given in the film flow regime. The interfacial mass transfer is obtained from the energy jump condition of the interfacial heat transfer. The wall heat transfer package consists of a library of heat transfer correlations and a selection logic algorithm, producing a continuous boiling curve. The wall friction coefficient is calculated based on Reynolds number, and the form loss is specified at each node by users.

2.3 Nodalization methodology

MEDUSA provides flexible noding system for both the hydraulic and the heat structure meshes. A channel is a vertical or horizontal stack of mesh cells. A channel may represent a vertical or horizontal flow path appropriate to the geometry being modeled. Transverse connections called gaps can be specified between the channels. All regions composed of channels of the same vertical length and beginning at the same level are grouped together and referred to as a section. The sections are joined together to form the complete component mesh by specifying connections to the channels in adjacent sections at the top and bottom of each channels. Solid structures can be represented as cylindrical rods, hollow tubes or flat walls composed of any number of different materials with specified thermal properties.

2.4 Component models

Several specific models has been developed to simulate the hydraulic components of nuclear reactor system, including pumps, valves, and steam generator separator. The valve models includes pressurizer safety valve(PSV), main steam safety valve(MSSV), atmospheric dump valve(ADV), main steam isolation valve(MSIV), and turbine stop valve(TSV), etc. Break flow model consists of subcooled liquid, two phase, and superheated vapor critical flow. The critical flow model can be applied not only to pipe break, but also valve flow such as MSSVs, PSVs, and ADVs. Pump head and torque are determined from the pump speed and flow, based on the homologous curve given by user. The pump speed can be directly controlled by user input, or calculated from the balance equation between the motor torque and hydraulic torque, depending on user specified option. Generalized control units such as sum, proportional, integral, differential, lead, lag, etc., and latched or non-latched trip units are provided. Users can model any control logic using the generalized control and trip units.

3. Application to feed line break analysis

3.1 Initial conditions and major assumptions

For the comparison purpose, the same initial conditions are applied for both the MEDUSA and CESEC-III analyses. Initial core power is 2815 Mwt. The initial pressurizer pressure, reactor coolant flow rate, core inlet temperature, pressurizer level, steam generator level, feed water enthalpy are at full power steady state condition. The set points of high pressurizer pressure reactor trip and the pressurizer safety valve are assumed to be 2,460 psia and 2,500 psia, respectively. Credit is not taken for the pressurizer level control system, and pressurizer pressure control system. A 0.2 ft^2 crack in the main feedwater line is assumed to instantaneously terminate feedwater flow to both steam generators. Loss of both normal onsite and offsite electric power is assumed to occur at the time of reactor trip.

3.2 Results

Fig. 1 shows the comparison between the break flows predicted by CESEC-III and MEDUSA. In CESEC-III analysis, the break flow rate is nearly constant for about 40 seconds, since saturated liquid is assumed to discharge from the break until the affected steam generator is depleted. However, it can be seen that MEDUSA predicts more realistic transition of break flow between subcooled liquid flow, two phase flow, and single phase vapor critical flow. The realistic transition of break flow results in relatively small break flow. Consequently, the level depletion time of the affected steam generator is delayed in MEDUSA analysis. Fig. 2 shows the variations of pressurizer pressure. The absence of subcooled water and the steam generator level depletion cause pressurizer pressure increase. As a result, the pressurizer pressure reaches the reactor trip set point at around 40 seconds. The relatively delayed reactor trip predicted in the MEDUSA analysis seems due to the delayed steam generator level depletion. Other major phenomena such as secondary pressure, reactor coolant pump coast down, PSV flow show a good agreement between CESEC-III and MEDUSA analyses.

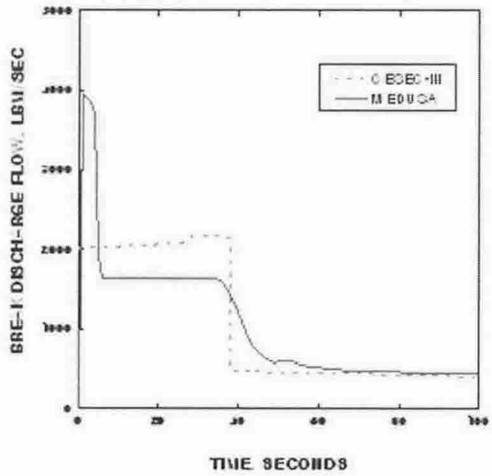

Fig. 1 Break discharge flow

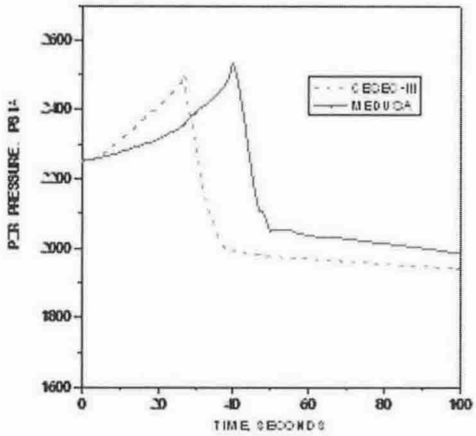

Fig. 2 Pressurizer pressure

4. Conclusion

The MEDUSA code has been developed, and applied to feed line break analysis for KSNP. It is demonstrated to model successfully the primary and secondary system of a nuclear power plant. The application results show that MEDUSA prediction agrees well with the results of the licensing code, CESEC-III, except the more realistic simulation of break flow and its direct impact on system parameters. Hence the general purpose thermal hydraulic analysis code, MEDUSA, is evaluated to be applicable to a typical non-LOCA accident, FLB analysis. Moreover, the MEDUSA code is expected to be useful to find additional safety margin, with more realistic simulation of two phase flow and the relevant phenomena.

REFERENCES

[1] M. J. Thurgood, et. al., "COBRA/TRAC - A Thermal-Hydraulics Code for Transient Analysis of Nuclear Reactor Vessels and Primary Coolant Systems". NUREG/CR-3046, 1982.

[2] C. E. Park, MEDUSA and its Application to Non-LOCA Analysis, 2nd Safety Analysis Symposium, 2004.