PSA METHODOLOGY DEVELOPMENT AND APPLICATION IN JAPAN

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ABSTRACT

The outlines of Japanese activities on development and application of probabilistic safety assessment (PSA) methodologies are described. First the activities on methodology development are described for system reliability analysis, operational data analysis, core melt accident analysis, environmental consequence analysis and seismic risk analysis. Then the methodology application examples by the regulatory side and the industry side are described.

1. INTRODUCTION

The probabilistic safety assessment (PSA) methodology has a capability to analyse and evaluate the frequency, progression and consequence of all the possible accident sequences at nuclear facilities. With this methodology, the overall safety of the different kinds of facilities is quantified using a single, common measure risk . Due to this characteristic, the PSA methodology is recognized as one of the best tools for discussing the balanced design and regulation of the facilities.

In the PSA of nuclear power plants, a special attention is paid to the accident scenarios resulting in severe core damage. This is because it has been recognized that only such severe accidents dominate the risk of the power plants since the Reactor Safety Study (RSS) was published in 1975. This recognition was strengthened through the accident at Three Mile Island in 1978 and the accident at Chernobyl in 1986. Thus the PSA of power plants can be defined as the evaluation of risk for the possible accident scenarios that may result in damage of the core.

In Japan, the usefulness of the PSA has been more and more widely recognized during recent several years. Governmental agencies and research institutes have been interested in the development of methodologies, aiming at their future usage in regulation. On the other hand, industry groups have paid more attention to its application to practical problems. In this report, the authors want to introduce the outlines of such development and application status of the PSA methodology in Japan.

2. METHODOLOGY DEVELOPMENT

In Japan several organizations including the Japan Atomic Energy Research Institute (JAERI) are developing the methodologies for PSA. The activities on the PSA methodology development can be categorized as follows:

(1) Methodology development for reliability analysis
(2) Collection and analysis of operational data
(3) Methodology development for core melt accident analysis
(4) Methodology development for environmental consequence analysis
(5) Methodology development for external event analysis

For each of these categories, outlines of Japanese status will be described, with a special emphasis on the JAERI's activities.

The program of PSA Methodology Development at JAERI is carried out according to the Five Year Safety Research Plan in Japan. The schedule of the program is illustrated in Fig. 1. As shown in the figure, it was started in 1980 for internal events and expanded in 1985 so as to include external events.

2.1. Analysis of System Reliability

At JAERI, there have been provided code packages for system reliability analysis based on two different methods - fault tree analysis method and GO method. As for the fault tree analysis method, the FTA-J code package was first developed[1] by combining several computer codes developed in the USA, aiming at a through analysis of system reliability. Then a new code package, REFT, was developed, most of those member codes were JAERI original. The REFT code package is capable of identification of minimal cut set, point estimation of system unavailability, evaluation of its uncertainty, and drawing of fault trees. It also has a capability to develop fault trees by combining modular trees in an interactive mode with the computer. As for the GO method, the GO-UA code package was developed at JAERI by modifying the GO code developed at EPRI. It has a capability to perform an uncertainty analysis which was not included in the original one. Since REFT and GO-UA are based on quite different methodologies (Former is inductive and latter is deductive.), the credibility of system reliability analysis can be examined by comparing the results with both code packages.

The codes mentioned above have been utilized for the reliability analyses for many of the safety-related systems of the Browns Ferry Unit 1 (BWR). The system reliability methodologies have been upgraded reflecting the experiences obtained through these analyses.

For the reliability analysis of large-scale and complicated systems including support systems, a new method - decomposition and integration method (D/I Method) - was developed[2]. The overall procedure is illustrated in Fig. 2. The method enabled us to perform practical and efficient analysis of a large-scale system, by decomposing the system into several subsystems and integrating the results obtained through the reliability analyses for individual subsystems.

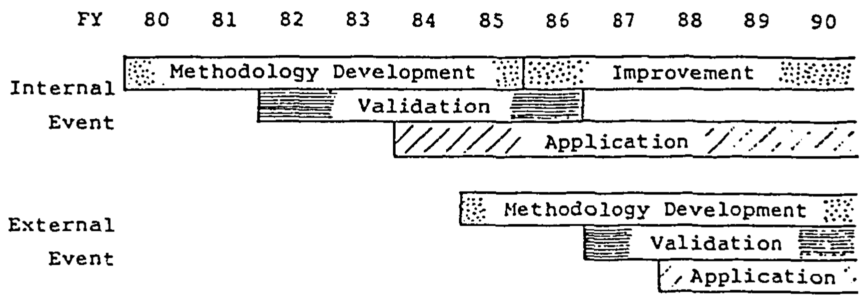

Fig.1 Milestone Chart of PSA Methodology Development Program at JAERI

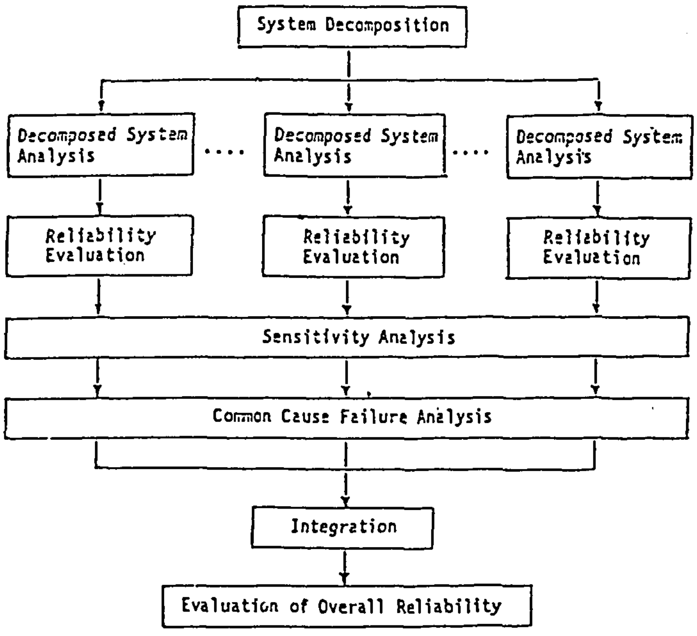

Fig.2 Procedure of D/I Method .

As for human reliability analysis, a new methodology DeBDA (Detailed Block Diagram Analysis) was developed[3]. The methodology was applied to practical problems and the results were incorporated into a system reliability analysis.

Besides these activities at JAERI, several research and development programs have progressed in Japan. The Power Reactor and Nuclear Fuel Development Corporation (PNC) and NSSS venders established their own methodologies for system reliability analysis, mostly based on the fault tree method. As for GO method, Ship Research Institute developed the GO-FLOW code[4] which is another modified version of the GO code.

2.2. Collection and Analysis of Operational Data

In order to support the system reliability analysis, component reliability data are required. In Japan, electric utilities and NSSS venders are collecting component reliability data through their operation experiences. These data are sent to the Central Research Institute of Electric Power (CRIEPI) and the Nuclear Power Engineering Test Center (NUPEC) and analysed and evaluated there. Human reliability data are collected by utilities at the BWR and the PWR training centers using plant simulators.

At JAERI, the computerized component failure rate database, RECORD, was developed. About 10,000 failure rate data were collected from about fifteen data sources published outside of Japan and codified for easy usage.

As for FBR, PNC is collecting the reliability data of its experimental reactor Monju as well as its test facilities. The PNC also has a program for exchanging these data with those in CREDO database in USA.

Abnormal occurrence data are not only analysed by the relevant utilities but also sent to and analysed by CRIEPI and NUPEC. CRIEPI is also exchanging these data with foreign utilities by joining the SEE-IN System, which is the Significant Event Evaluation and Information Network System operated by the Institute of Nuclear Power Operations (INPO) in USA.

Of the data submitted from utilities to the Ministry of International Trade and Industry (MITI) the significant ones are sent to OECD/NEA's Incident Reporting System (IRS) which aims at international exchange and mutual usage of the data on rare and significant events. Receiving the IRS data from NEA, JAERI is analysing and evaluating the important events. For supporting this activity, JAERI has developed some tools, including the CESAS software[5] which is to abstract the causal and sequential relationships between the events which occured in the course of an accident automatically with the aid of the computer.

2.3. Analysis of Core Melt Accident

The core melt analysis is for evaluating so called source terms' which are quantities and chemical forms of the fission products released into the environment following the variety of accident scenarios. In order to estimate the source terms, JAERI developed the THALES/ART code package[6], which consists of THALES for analysing progression of a core melt accident and ART for fission product release and transport behavior under the accident conditions.

The THALES/ART code package has a lot of special features comparing to the USNRC's Source Term Code Package[7]. For example, the reactor coolant system of PWR or BWR is divided into multiple volumes, molten fuel is relocated downward, voidage of water is dealt with also in the containment, and fission product transport carried even by liquid flow is analysed. With these models it has become possible to treat some of the phenomena which couldn't be analysed with the USNRC's code package, including (1) Automatic actuation and termination of HPCI and RCIC reflecting the change in the water level in the reactor pressure vessel (in the case of BWR), (2) Fission product scrubbing in the pressurizer when water is retained there (in the case of transient sequences of PWR), (3) Change in hydrogen generation rate and fission product release rate reflecting the change in molten node slumping velocity, and (4) Flashing of water in the sump or suppression pool when the containment fails and is depressurized.

In order to validate the THALES/ART code package, analyses of experiments and comparison with other codes have been conducted. For example, fuel rod thermal behavior during the Severe Fuel Damage (SFD) tests at the Power Burst Facility (PBF) was analysed with the THALES code[8] and aerosol sedimentation behavior during the Nuclear Safety Pilot Plant (NSPP) tests at the Oak Ridge National Laboratory (ORNL) was analysed with the ART code. Most of the results of these analyses showed a good agreement with the measured data and the validation of the codes has progressed. As an example, Fig. 3 shows the measured and calculated fuel temperature for the SFD Scoping Test.

With the THALES/ART code package, sensitivity analyses were carried out for the in-vessel accident progression, ex-vessel accident progression and fission product release and transport for both of the model PWR (Indian Point 3) and the model BWR (Browns Ferry 1).

The computed results from one of the sensitivity analyses for PWR are shown in Fig. 4[9]. The sequence assumed in this run is a small break LOCA with the failure of ECCS and containment spray. The uppermost column shows the corium temperature and the containment pressure. The middle shows the cesium iodide generation rate. (In this analysis, it was assumed that all the released iodine was transformed into cesium iodide.) The lowermost shows the change in the cesium iodide distribution. The results of this analysis show that some amount of iodine is released from the corium in the reactor cavity after the containment failure and this later release dominates the iodine release to the environment.

Through these analyses, the mechanisms which govern the accident progression and fission product release and transport were made clearer. Some conclusions were similar to those of the source term studies conducted by the USNRC[10] and the Industry Degraded Core Rule-making (IDCOR) Group[11]. For example, it was confirmed that the source terms would be strongly influenced by assumed accident sequences and plant configurations and that in-vessel fission product retension could not be neglected in most sequences. On the other hand, some new results were also obtained. For example, the multivolume thermal hydraulics model and the molten fuel relocation model were found to have significant effects on the source terms.

In order to support the development and validation of the THALES/ART code package, JAERI is continuing several experimental studies and detailed code development. For example, fuel meltdown tests are carried out with the Nuclear Safety Research Reactor (NSRR) and debris coolability tests are carried out with an out-of-pile test facility.

Besides these JAERI activities, utilities are using and modifying the MAAP code which was developed in the IDCOR Program. As for FBR, PNC is developing a code package for analysing a hypothetical core damage accident.

2.4. Analysis of Environmental Consequence

In order for assessing the off-site radiological consequence of nuclear accidents, the OSCAAR computer code package is being developed at JAERI. As shown in Fig. 5, OSCAAR consists of a series of interlinked computer codes which are used to predict (1) Transport of radionuclides through the environment to man, (2) Subsequent dose distributions, and (3) Health effects in the population.

Up to now, two codes have been developed for the radionuclide transport and dose distribution analysis. One is for calculating early exposure which occurs during and shortly after plume passage and the other is for chronic exposure due to radionuclides remaining in the environment for a long period. The computer code to evaluate the health effects will be developed in the near future.

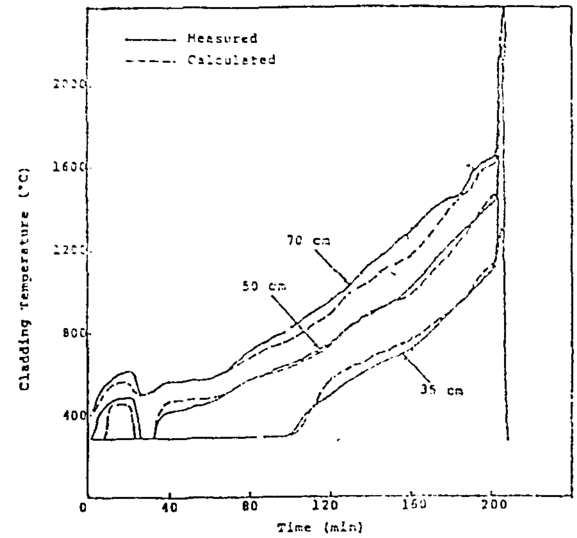

Fig.3 Thermal Response Analysis of PBF/SFD Scoping Test with THALES-M Code

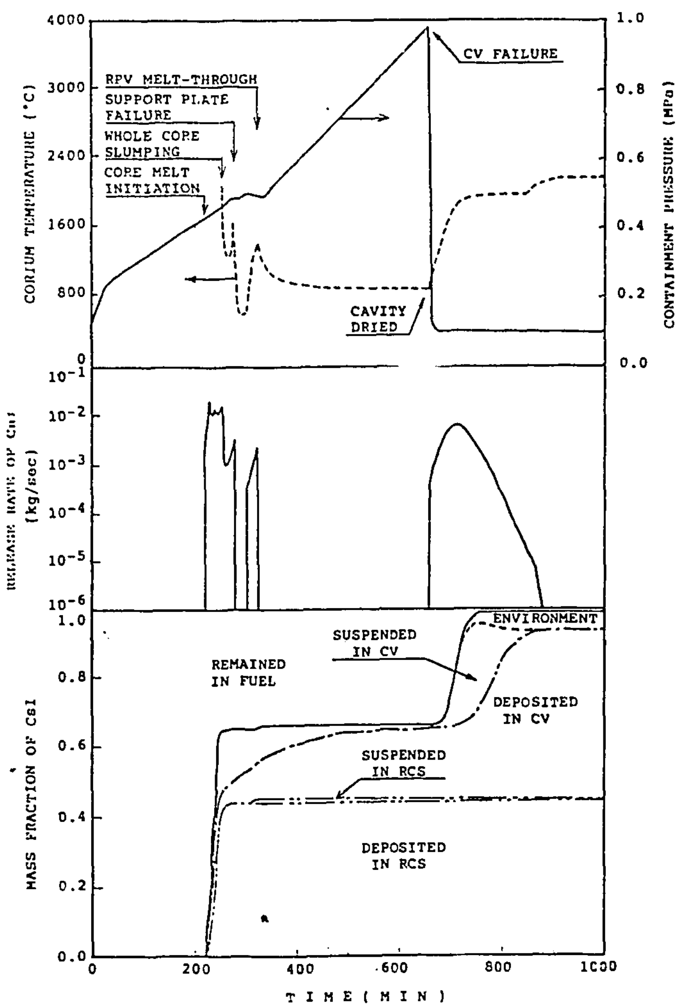

Fig.4 Source Term Evaluation with THALES/ART Code Package (Small Break LOCA with ECCS Failure)

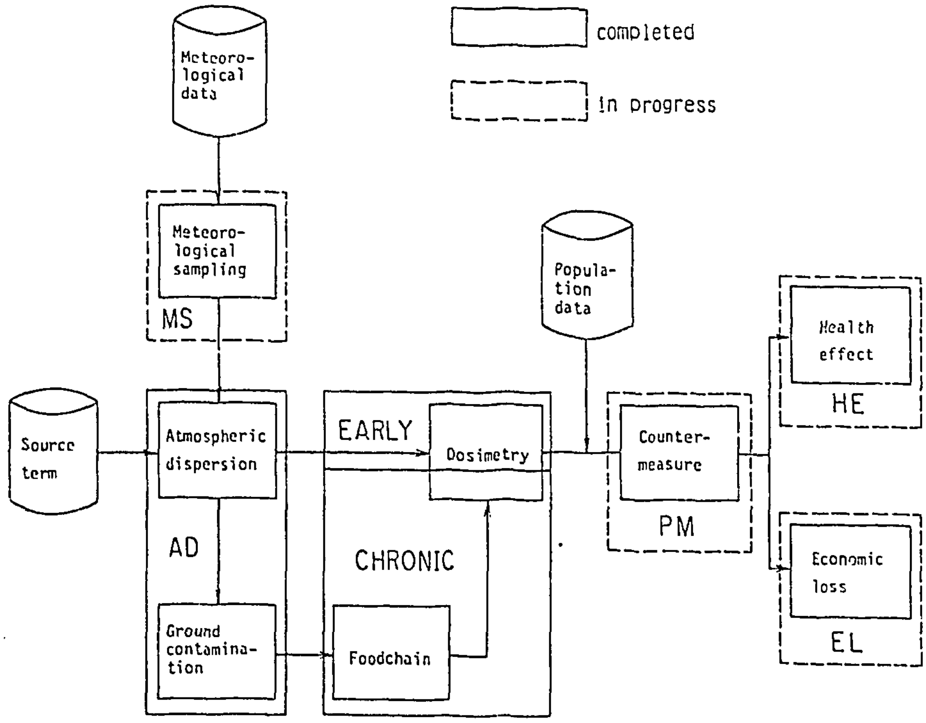

Fig.5 Structure of the OSCAAR Code System

2.5. Analysis of Seismic Risk

In Japan, JAERI, utilities and general construction companies are now eager to developing the methodologies for seismic risk analysis, intending to use the analysis results in seeking more balanced seismic designs or regulations. The general procedure for the seismic risk analysis consists of (1) Seismic hazard evaluation, (2) Response evaluation, (3) Component failure probability evaluation, and (4) System analysis and accident frequency evaluation.

Of these tasks for the seismic risk analysis, the seismic hazard evaluation has progressed in many of the organizations in Japan. In these evaluations, long history of earthquake records were used as well as active fault data, of course with interpretations by experts.

At JAERI, a prototype code named SHEAT was developed and used for the sensitivity analyses for two sites in Japan. Through these analyses, the parameters which will cause large uncertainty in seismic hazard were identified to be (1) Interpretation of seismic sources, (2) Mathematical expression of the sources, and (3) Selection of attenuation correlation.

In succession to the methodology development for the hazard evaluation, the methogologies to cover all the required tasks are under development in some organizations including JAERI.

3. METHODOLOGY APPLICATION

3.1 Licensing Application

Most of current licensing procedures in Japan are based on deterministic approach with a few exceptions. In licensing application, the probabilistic approach has been used to supplement the regulatory decision, but will not replace the deterministic approach. A few examples of applying the probabilistic approach to regulatory decisions are presented as follows:

a) The air-crash accident and the turbine missile accident are assessed for each plant under licensing with a direct application of probabilistic method. For the air-crash accident, no provisions are required if the probability of air plane crash to a nuclear power plant is less than 10^{-7}/R.Y. For the turbine missile accident, the guideline requires to prove the combined probabilities, probabilities of occurrence of turbine missile, missile collision to an essential part of the plant and its failure occurrence, be less than 10^{-7} /R.Y.

b) The following examples are of more generic application carried out:

Station blank out

In the current LWR safety design guideline (1977), it is required that safe shutdown should be ensured in case of a total loss of AC power supply to a nuclear power plant for a certain short time, which is in practice to be understood as about 30 minutes. Generic study was carried out to evaluate the reliability of the power grid and the on-site emergency power supply, i.e. diesel generators. The high voltage transmissions in Japan have shown quite reliable and its failure probability for longer than 30 minutes is about 10^{-4} /km-year by the statistics. The emergency diesel generators have been tested regularly at the sites and the unavailability was revealed to be less than 1.2×10^{-3} /demand, notabaly no failure reported in recent years. The study concludes that the probability of a station black out for a significant long time seems sufficiently low and that it could be precluded from the design basis events.

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The anticipated transient without scram is considered to be beyond design basis event in Japan. For this decision, the failure probability of reactor shutdown systems for our domestic reactors was assessed by the statistics obtained from our operating experience and fault tree analysis, which was found to be far less than 10^{-6} /R.Y.

Multiple Failures of Process Systems

Through operating experiences, a question was raised if any failures of process systems (which are non-safety-related, not directly involved in a certain safety function and not necessarily redundant) would cause any problem. Many cases were examined with use of event charts for various failure modes and their consequences as well as their postulated frequencies of occurrence. The study concluded that the failures of these systems, even combined with a single failure of the redundant safety systems, were either of very low probability or of not significant consequences.

c) The followings are other topics under discussion:

The current reactor siting guideline (1964) requires that the reactor shall be adequately separated from the surrounding society with three binds of distances based on different accident scenarios and different acceptance criteria. Two different scenarios are analysed, but it has not been clear enough why two accidents are selected to be analysed. A proposal is now raised that the probability of any non-stochastic health detriments of the public by accidental radiation exposure should be kept lower than an acceptable value, e.g. 10^{-6} /R.Y. The site evaluation accident scenario would be fixed to suffice the proposal. Now risk dominant sequences are being collected and evaluated with a consideration of our demestic operating experiences and of results of current source term studies. In these discussions, the integrity of the containment is identified to be of key importance. A group of experts has been set up to discuss the ultimate role of the containment namely to what extent the containment failure probability be kept low. The necessity fixing a new DBE for the containment and accident/risk management of current designed containments is now discussed. This discussion will be accelerated by a impact of the Chelnobyl accident.

3.2 Application by Utilities and NSSS Venders

Electric utilities and NSSS venders are applying the PSA methodologies to various problems. For example, the Tokyo Electric Power Co. Ltd. (TEPCO) used the event tree technique in the design of an advanced BWR (A-BWR). NSSS venders are eager to using the technique of system reliability analysis in system design and plant maintenance modification. In this section, the outlines of the A-BWR design assessment by TEPCO will be discussed.

In their assessment, they in principle did not evaluate the system unavailability of the A-BWR based on their own system models and reliability data, but used the unavailability for each train of the safety-related systems evaluated for the Limerick plant in USA. This is because their objectives were to find out a good safety design for the A-BWR. By using the common unavailability data for the trains, they evaluated the differences in the probability of the core melt accident sequences which are due to the differences in the plant configurations.

The results of the assessment are illustrated in Fig. 6. As shown in the figure, the proposed plant configuration for the A-BWR will have a lower core melt probability comparing to those of Limerick and a current BWR-5 plant.
TEPCO also did a case study for this assessment, where the reliability data of Japanese power plants were partially used. Reflecting the lower initiating event frequency and the lower diesel generator failure probability at Japanese plants, the core melt probability of the A-BWR was much lower than that in the above assessment.

4. SUMMARY

In Japan, usefulness of PSA has been widely recognized and hence its methodology development has been continued. JAERI has been developing a set of methodologies for system reliability analysis, operational data analysis, core melt accident analysis, environmental consequence analysis and seismic risk analysis Reliability data are collected and evaluated mainly by industry groups. PSA methodologies for FBR are under development at PNC.

Although PSA will not directly be required in licensing process in Japan at least in the near future, probabilistic approach will be more widely utilized in regulation. For example, PSA results will be referred in making or reviewing regulatory rules. On the other hand, Japanese utilities and NSSS venders are applying the PSA methodlogies in design and maintenance modification.

After all, the PSA methodology development will be further continued and its application will be more popular in Japan.

References

1) T. Ishigami, et al., "User's Manual for the FTA-J (Fault Tree Analysis - JAERI) Code System", JAERI-M 83-169, (1983)
2) N. Watanabe, et al., "Large-Scale Reliability Analysis by Decomposition and Integration", Proc. Int. ANS/ENS Topical Mtg. on Probabilisitic Safety Method and Applications, San Francisco, (1985)
3) T. Yukimachi, et al., "A Method for Quantitative Reliability Analysis of Man-Machine Systems", Human Engineering, Vol.18, No.12, (1982)
4) T. Matsuoka, et al., "GO-FLOW; A Reliability Analysis Methodology Applicable to Piping System", Ref. 2)
5) N. Watanabe, et al., "Development of the Computerized Event Sequence Abstracting System", Int. ANS/ENS Topical Mtg. on Thermal Reactor Safety, San Diego, (1986)
6) K. Abe, et al., "Overview of Development and Application of THALES Code System for Analyzing Progression of Core Meltdown Accident of LWR's", Proc. 2nd Int. Topical Mtg. on Nuclear Power Plant Thermal Hydraulics and Operations, Tokyo, (1986)
7) T. S. Kress, "Review of the Status of Validation of the Computer Codes Used in the Severe Accident Source Term Study (BMI-2104), ORNL/TM-8842, (1985)
8) K. Abe et al., "A Study on the Most Influential Parameters on the Progression of PWR Core Meltdown Accident", Ref.5
9) K. Abe et al., "Effects of In-Vessel Thermal-Hydraulics Models on Source Terms", Int. SNS/ENS/ANS Topical Mtg. on Probabilistic Safety Assessment and Risk Management, (To be Published, 1987)
10) U.S.NRC, "Reassessment of the Technical Bases for Estimating Source Terms" NUREG-0956, (1985)
11) M. H. Fontana, et al., "The IDCOR Programme; Severe Accident Issues, Individual Plant Examinations and Source Term Development", Proc. IAEA Symp. on Source Term Evaluation for Accident Conditions, Knoxville, (1985)

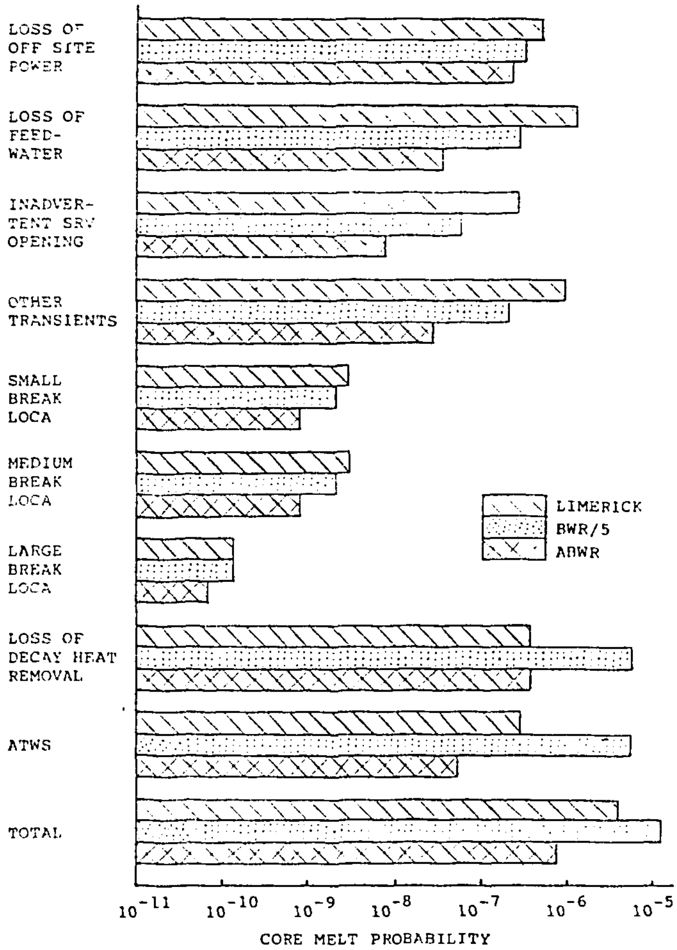

Fig.6 Comparison of Core Melt Probabilities