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Kalinin VVER-1000 Nuclear Power Station Unit 1 PRA

Procedure Guides for a Probabilistic Risk Assessment

English Version

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ABSTRACT

In order to facilitate the probabilistic risk assessment (PRA) of a VVER-1000 nuclear power plant, a set of procedure guides has been written. These procedure guides, along with training supplied by experts and supplementary material from the literature, were used to advance the PRA carried out for the Kalinin Nuclear Power Station in the Russian Federation. Although written for a specific project, these guides have general applicability. Guides are procedures for all the technical tasks of a Level 1 (determination of core damage frequency for different accident scenarios), Level 2 (probabilistic accident progression and source term analysis), and Level 3 (consequence analysis and integrated risk assessment) PRA. In addition, introductory material is provided to explain the rationale and approach for a PRA. Procedure guides are also provided on the documentation requirements.

FOREWORD

During the Lisbon Conference on Assistance to the Nuclear Safety Initiative, held in May 1992, participants agreed that efforts should be undertaken to improve the safety of nuclear power plants that were designed and built by the former Soviet Union. That agreement led to a collaborative probabilistic risk assessment (PRA) of the Kalinin Nuclear Power Station (KNPS), Unit 1, in the Russian Federation. The KNPS Unit 1 PRA was intended to demonstrate the benefits obtained from application of risk technology towards understanding and improving reactor safety and, thereby, helping to build a risk-informed framework to help address reactor safety issues in regulations.

The U.S. Department of State, together with the Agency for International Development (AID), requested that the U.S. Nuclear Regulatory Commission (NRC) and the Federal Nuclear and Radiation Safety Authority of the Russian Federation (Gosatomnadzor, or GAN) work together to begin applying PRA technology to Soviet-designed plants.[1] On the basis of that request, in 1995, the NRC and GAN agreed to work together to perform a PRA of a VVER-1000 PWR reactor. Under that agreement, the NRC provided financial support for the PRA with funds from AID and technical support primarily from Brookhaven National Laboratory and its subcontractors. KNPS Unit 1 was chosen for the PRA, and the effort was performed under the direction of GAN with the assistance of KNPS personnel and the following four other Russian organizations:

- Science and Engineering Centre for Nuclear and Radiation Safety (GAN's and now Rostechnadzor's technical support organization)
- Gidropress Experimental and Design Office (the VVER designer)
- Nizhny Novgorod Project Institute, "Atomenergoprojekt" (the architect-engineer)
- Rosenergoatom Consortium (the utility owner of KNPS)

One of the overriding accomplishments of the project has been technology transfer. In NRC-sponsored workshops held in Washington, DC, and Moscow from October 1995 through November 2003, training was provided in all facets of PRA practice. In addition, the Russian participants developed expertise using current-generation NRC-developed computer codes, MELCOR, SAPHIRE and MACCS. Towards the completion of the PRA, senior members of the Kalinin project team began the development of risk-informed, Russian nuclear regulatory guidelines. These guidelines foster the application of risk assessment concepts to promote a better understanding of risk contributors. Efforts such as this have benefited from the expertise obtained, in part, from the training, experience, and insights gained from participation in the KNPS Unit 1 PRA project.

The documentation of the Kalinin PRA comprises two companion NUREG-series reports:

- NUREG/CR-6572, Revision 1, "Kalinin VVER-1000 Nuclear Power Station Unit 1 PRA: Procedure Guides for a Probabilistic Risk Assessment," was prepared by Brookhaven National Laboratory and the NRC staff. It contains guidance for conducting the Level 1, 2, and 3 PRAs for KNPS with primary focus on internal events. It may also serve as a guide for future PRAs in support of other nuclear power plants.

[1]As a result of a governmental decree in May 2004, GAN was subsumed into a new organization, known as the Federal Environmental, Industrial and Nuclear Supervision Service of Russia (Rostechnadzor).

- NUREG/IA-0212, "Kalinin VVER-1000 Nuclear Power Station Unit 1 PRA: Volumes 1 and 2," was written by the Russian team and, by agreement, includes both a non-proprietary and proprietary volume. The non-proprietary volume, Volume 1, "Executive Summary Report," discusses the project objectives, summarizes how the project was carried out, and presents a general summary of the PRA results. The proprietary volume, Volume 2, contains three parts. Part 1, "Main Report: Level 1 PRA, Internal Initiators," discusses the Level 1 portion of the PRA; Part 2, "Main Report: Level 2 PRA, Internal Initiators," discusses the Level 2 portion; and Part 3, "Main Report: Other Events Analysis," discusses preliminary analyses of fire, internal flooding, and seismic events, which may form the basis for additional risk assessment work at some future time.

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3.3.5.5 References

EPRI, "MAAP4 - Modular Accident Analysis Program for LWR Power Plants," RP3131-02, Volumes 1-4, Electric Power Research Institute, 1994.

Summers, R., M, et al., "MELCOR Computer Code Manuals -- Version 1.8.3," NUREG/CR-6119, SAND93-2185, Volumes 1-2, Sandia National Laboratories, 1994.

NRC, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, U.S. Nuclear Regulatory Commission, December 1990.

Harper, F. T., et al, "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters," NUREG/CR-4551, Volume 2, SAND86-1309, Sandia National Laboratories, December 1990.

NRC, "Individual Plant Examination: Submittal Guidance," NUREG-1335, U.S. Nuclear Regulatory Commission, August 1989.

Jow, H. J., et al., "XSOR Codes User Manual," NUREG/CR-5360, Sandia National Laboratories, 1993.

3.4 Level 3 Analysis (Consequence Analysis and Integrated Risk Assessment)

In this section, the analyses performed as part of the Level 3 portion of a probabilistic risk assessment (PRA) are described.

3.4.1 Assumptions and Limitations

In most Level 3 (i.e., consequence) codes, atmospheric transport of the released material is carried out assuming Gaussian plume dispersion. This assumption is generally valid for flat terrain to a distance of a few kilometers from the point of release but is inaccurate both in the immediate vicinity of the reactor building and at farther distances. For most PRA applications, however, the inaccuracies introduced by the assumption of Gaussian plumes are much smaller than the uncertainties due to other factors, such as the source term. In specific cases of plant location, such as, for example, a mountainous area or a valley, more detailed dispersion models that incorporate terrain effects may have to be considered. There are other physical parameters that influence downwind concentrations. Dry deposition velocity can vary over a wide range depending on the particle size distribution of the released material, the surface roughness of the terrain, and other factors. An assessment of these uncertainties focused on the factors which influence dispersion and deposition has been carried out recently (Harper et al., 1995). Earlier assessments of the assumptions and uncertainties in consequence modeling were reported in other PRA procedures guides (NRC, 1983).

Besides atmospheric transport, dispersion, and deposition of released material, there are several other assumptions, limitations, and uncertainties embodied in the parameters that impact consequence estimation. These include: models of the weathering and resuspension of material deposited on the ground, modeling of the ingestion pathway, i.e., the food chains, ground-crop-man and ground-crop-animal-dairy/meat-man, internal and external dosimetry, and the health effects model parameters. Other sources of uncertainty arise from the assumed values of parameters that determine the effectiveness of emergency response, such as the shielding provided by the building stock in the area where people are assumed to shelter, the speed of evacuation, etc. Comparison of the results of different consequence codes, which embody different approaches and values of these parameters, on a standard problem are contained in a study sponsored by the Organization for Economic Co-operation and Development (OECD, 1994). An uncertainty analysis of the COSYMA code results using the expert elicitation method is currently being carried out (Jones, 1996).

3.4.2 Products

Documentation of the analyses performed to estimate the consequences associated with the accidental release of radioactivity to the environment should contain sufficient information to allow an independent analyst to reproduce the results. At a minimum, the following information should be documented for the Level 3 analysis:

- identification of the consequence code and the version used to carry out the analysis,
- a description of the site-specific data and assumptions used in the input to the code,
- specifications of the source terms used to run the code, and
- discussion and definition of the emergency response parameters,
- a description of the computational process used to integrate the entire PRA model (Level 1 - Level 3),
- a summary of all calculated results including frequency distributions for each risk measure.

3.4.3 Analytical Tasks

A Level 3 PRA consists of two major tasks:

1. Consequence analyses conditional on various release mechanisms (source terms) and

2. Computation of risk by integrating the results of Levels 1, 2, and 3 analyses.

Task 1 – Consequence Analysis

The consequences of an accidental release of radioactivity from a nuclear power plant to the surrounding environment can be expressed in several ways: impact on human health, impact on the environment, and impact on the economy. The consequence measures of most interest to a Level 3 PRA focus on the impact to human health. They should include:

- number of early fatalities,

- number of early injuries,

- number of latent cancer fatalities,

- population dose (person-rem or person-sievert) out to various distances from the plant,

- individual early fatality risk defined in the early fatality QHO, i.e., the risk of early fatality for the average individual within 1 mile from the plant, and

- individual latent cancer fatality risk defined in the latent cancer QHO, i.e., the risk of latent cancer fatality for the average individual within 10 miles of the plant.

The consequence measures that focus on impacts to the environment include:

- land contamination

- surface water body (e.g., lakes, rivers, etc.) contamination.

Groundwater contamination has yet to be included in a Level 3 analyses, although it may be important to consider it in certain specific cases.

The economic impacts are mainly estimated in terms of the costs of countermeasures taken to protect the population in the vicinity of the plant. These costs can include:

- short-term costs incurred in the evacuation and relocation of people during the emergency phase following the accident and in the destruction of contaminated food, and

- long-term costs of interdicting contaminated farmland and residential/urban property which cannot be decontaminated in a cost-effective manner, i.e., where the cost of decontamination is greater than the value of the property.

The costs of medical treatment to potential accident victims are not generally estimated in a Level 3 analysis, although approaches do exist for incorporating these costs (Mubayi, 1995) if required by the application.

The results of the calculations for each consequence measure are usually reported as a complementary cumulative distribution function. They can also be reported in terms of a distribution--for example, ones that show the 5th percentile, the 95th percentile, the median, and the mean.

A probabilistic consequence assessment (PCA) code is needed to perform the Level 3 analysis. Such codes normally take as input the characteristics of the release or source term provided by the Level 2 analysis. These characteristics typically include for each specified source term: the release fractions of the core inventory of key radionuclides, the timing and duration of the release, the height of the release (i.e., whether the release is elevated or ground level), and the energy of the release. PCA codes incorporate algorithms for performing weather sampling on the plume transport in order to obtain

a distribution of the concentrations and dosimetry which reflect the uncertainty and/or variability due to weather. The codes also model various protective action countermeasures to permit a more realistic calculation of doses and health effects and to assess the efficacy of these different actions in reducing consequences.

Several PCA codes are currently in use for calculating the consequences of postulated radiological releases. The NRC supports the use of the MACCS (Jow, 1990 and Chanin, 1993) and MACCS2 (Chanin and Young, 1997) PCA codes for carrying out nuclear power plant Level 3 PRA analyses. A number of countries in Europe support the use of the COSYMA (KfK and NRPB, 1991 and Jones, 1996) PCA code for their Level 3 analyses.

PCA codes require a substantial amount of information on the local meteorology, demography, land use, crops grown in various seasons, foods consumed, and property values. For example, the input file for the MACCS code requires the following information:

- Meteorology - one year of hourly data on: windspeed and direction, atmospheric stability class, precipitation rate, probability of precipitation occurring at specified distances from the plant site, and height of the atmospheric inversion layer.

- Demography - population distribution around the plant on a polar grid defined by 16 angular sectors and user-specified annular radial sectors, usually a finer grid close to the plant and one that becomes progressively coarser at greater distances.

- Land Use - fraction which is land, land which is agricultural, major crops, and growing season.

- Economic Data - value of farmland, value of nonfarm property, and annual farm sales.

The MACCS User Manual (Chanin, 1990) and the MACCS2 User Guide (Chanin and Young, 1997) may be consulted for a complete description of the site input data necessary.

In addition to site data, a PCA code should have provisions to model countermeasures to protect the public and provide a more realistic estimate of the doses and health effects following an accidental release. The MACCS code requires that the analyst make assumptions on the values of parameters related to the implementation of protective actions following an accident. The types of parameters involved in evaluating these actions include the following:

- delay time between the declaration of a general emergency and the initiation of an emergency response action, such as evacuation or sheltering; this delay time may be site specific,

- fraction of the offsite population which participates in the emergency response action,

- effective evacuation speed,

- degree of radiation shielding provided by the building stock in the area,

- projected dose limits for long-term relocation of the population from contaminated land, and

- projected ingestion dose limits used to interdict contaminated farmland.

The selected values assumed for the above (or similar) parameters need to be justified and documented since they have a significant impact on the consequence calculations.

In summary, the PCA code selected for the calculation of consequences should have the following capabilities:

- incorporate impact of weather variability on plume transport by performing stratified or Monte Carlo sampling on an annual set of relevant site meteorological data,

- allow for plume depletion due to dry and wet deposition mechanisms,

- allow for buoyancy rise of energetic releases,

- include all possible dose pathways, external and internal (such as cloudshine, groundshine, inhalation, resuspension inhalation, and ingestion) in the estimation of doses,

- employ validated health effects models based, for example, on (ICRP, 1991) or BEIR V (National Research Council, 1990) dose factors for converting radiation doses to early and latent health effects, and

- allow for the modeling of countermeasures to permit estimation of a more realistic impact of accidental releases.

The above-cited methods for estimating consequences are, in general, adequate for accidents caused by internal initiating events during both full power operation and shutdown conditions. However, for external initiating events, such as seismic events, certain changes may be needed. For example, the early warning systems and the road network may be disrupted so that initiation and execution of emergency response actions may not be possible. Hence, in addition to changing the potential source terms, a seismic event could also influence the ability of the close-in population to carry out an early evacuation. A Level 3 seismic PRA should, therefore, include consideration of the impacts of different levels of earthquake severity on the consequence assessment.

To use a consequence code, generally the following data elements are required:

- reactor radionuclide inventory,
- accident source terms defined by the release fractions of important radionuclide groups, the timing and duration of the release, and the energy and height of the release,
- hourly meteorological data at the site as recommended, for example, in Regulatory Guide 1.23 (NRC, 1986), collected over one or, preferably, more years and processed into a form usable by the chosen code,
- site population data from census or other reliable sources and processed in conformity with the requirements of the code, i.e., to provide population information for each area element on the grid used in the code,
- site economic and land use data, specifying the important crops in the area, value and extent of farm and nonfarm property,
- defining the emergency response countermeasures, including the possible time delay in initiating response after declaration of warning and the likely participation in the response by the offsite population.

Task 2 – Computation of Risk

The final step in a Level 3 PRA is the integration of results from all previous analyses to compute individual measures of risk. The severe accident progression and the radionuclide source term analyses conducted in the Level 2 portion of the PRA, as well as the consequence analysis conducted in the Level 3 portion of the PRA, are performed on a conditional basis. That is, the evaluations of alternative severe accident progressions, resulting source terms, and consequences are performed without regard to the absolute or relative frequency of the postulated accidents. The final computation of risk is the process by which each of these portions of the accident analysis are linked together in a self-consistent and statistically rigorous manner.

An important attribute by which the rigor of the process is likely to be judged is the ability to demonstrate traceability from a specific accident sequence through the relative likelihood of alternative severe accident progressions and measures of associated containment performance (i.e., early versus late failure) and ultimately to the distribution of fission product source terms and consequences. This traceability should be demonstrable in both directions, i.e., from the accident sequence to a distribution of consequences and from a specific level of accident consequences back to the fission product source terms, containment performance measures, or accident sequences that contribute to that consequence level.

3.4.4 Task Interfaces

The current task requires a set of release fractions (or source terms) from the Level 2 analysis (Section 3.3) as input to the consequence analysis.

The consequences are calculated in terms of: (1) the acute and chronic radiation doses from all pathways to the affected population around the plant, (2) the consequent health effects (such as early fatalities, early injuries, and latent cancer fatalities), (3) the integrated population dose to some specified distance (such as 50 miles) from the point of release, and (4) the contamination of land from the deposited material.

The consequence measures to be calculated depends on the application as defined in PRA

Scope. Generally, in a Level 3 analysis, a distribution of consequences is obtained by statistical sampling of the weather conditions at the site. Each set of consequences, however, is conditional on the characteristics of the release (or source term) which are evaluated in the Level 2 analysis.

An integrated risk assessment combines the results of the Levels 1, 2, and 3 analyses to compute the selected measures of risk in a self-consistent and statistically rigorous manner. The risk measures usually selected are: early fatalities, latent cancer fatalities, population dose, and quantitative health objectives (QHOs) of the U.S. Nuclear Regulatory Commission (NRC) Safety Goals (NRC, 1986). Again, the actual risk measures calculated will depend on the PRA Scope.

3.4.5 References

Chanin, D.I., and M. L. Young, "Code Manual for MACCS2: Volume 1, User's Guide," SAND97-0594, Sandia National Laboratories, March 1997.

Chanin, D.I., et al., "MACCS Version 1.5.11.1: A Maintenance Release of the Code," NUREG/CR-6059, Sandia National Laboratories, October 1993.

Chanin, D.I., et al., "MELCOR Accident Consequence Code System (MACCS), Volume 1, User's Guide," NUREG/CR-4691, Sandia National Laboratories, February 1990.

Harper, F. T., et al., "Probabilistic Accident Consequence Uncertainty Analysis, Dispersion, and Deposition Uncertainty Assessment," NUREG/CR-6244, Sandia National Laboratories, 1995.

ICRP, "1990 Recommendations of the ICRP," Annals of the ICRP, Vol. 21, No. 1-3, ICRP Publication 60, International Commission on Radiological Protection, Pergamon Press, Oxford, England, 1991.

Jones, J. A., et al., "Uncertainty Analysis on COSYMA," Proceedings of the Combined 3rd COSYMA Users Group and 2nd International MACCS Users Group Meeting, Portoroz, Slovenia, 41228-NUC 96-9238, KEMA, Arnhem, the Netherlands, September 16-19, 1996.

Jow, H. N., et al., "MELCOR Accident Consequence Code System (MACCS), Volume II, Model Description," NUREG/CR-4691, Sandia National Laboratories, February 1990.

KfK and NRPB, "COSYMA - A New Program Package for Accident Consequence Assessment," CEC Brussels, EUR 13028, Kernforschungszentrum (Karlsruhe) and National Radiological Protection Board, 1991.

Mubayi, V., et al., "Cost-Benefit Considerations in Regulatory Analysis," NUREG/CR-6395, Brookhaven National Laboratory, 1995.

National Research Council, "Health Effects of Exposure to Low Levels of Ionizing Radiation," BEIR V, Washington, DC, 1990.

NRC, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, Vol. 1, Main Report, U.S. Nuclear Regulatory Commission, 1990.

NRC, "Safety Goals for the Operation of Nuclear Power Plants, Policy Statement," Federal Register, Vol. 51, No. 149, U.S. Nuclear Regulatory Commission, August 4, 1986.

NRC, "Onsite Meteorological Programs," Regulatory Guide 1.23, U.S. Nuclear Regulatory Commission, April 1986.

NRC, "PRA Procedures Guide - A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," NUREG/CR-2300, Vol. 2, U.S. Nuclear Regulatory Commission, 1983.

OECD, "Probabilistic Accident Consequence Assessment Codes, Second International Comparison", Organisation for Economic Cooperation and Development, Nuclear Energy Agency, Paris, France, 1994.